JAERI-M
7034

最近の熱交換器に関する文献調査と HTGR用熱交換器の開発上の問題点

1977年4月

山尾　裕行*・岡本　芳三・佐野川　好母

日本原子力研究所
Japan Atomic Energy Research Institute

この報告書は，日本原子力研究所がJAERI-Mレポートとして，不定期に刊行している研究報告書です．入手，複製などのお問合わせは，日本原子力研究所技術情報部（茨城県那珂郡東海村）あて，お申しこしください．

JAERI-M reports, issued irregularly, describe the results of research works carried out in JAERI. Inquiries about the availability of reports and their reproduction should be addressed to Division of Technical Information, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken, Japan.

JAERI－M 7034

最近の熱交換器に関する文献調査とＨＴＧＲ用熱交換器の開発上の問題点

日本原子力研究所東海研究所原子炉工学部

山尾裕行*・岡本芳三・佐野川好母

（１９７７年2月２２日　受理）

１９６７年から１９７６年までにＮＳＡ（Nuclear Science Abstracts）に掲載された熱交換器に関する文献索引を利用し，その開発上の問題点を抽出し，検討を行った。文献を，炉型，熱交換器の型式，国別，冷却材，研究テーマの種類の各項目について分類した。その結果，炉型としては世界的にはＬＭＦＢＲとＨＴＧＲと軽および重水炉用の開発に重点が置かれていることが明らかになった。一方，ＨＴＧＲ用熱交換器の開発研究は，耐熱材料，腐食に関するテーマのものが，他の炉に比べて多くなされており，また，水素透過防止は，多目的高温ガス炉では重要な課題であることが認識された。

＊　三菱重工業株式会社よりの外来研究員

JAERI-M 7034

Bibliographical survey of heat exchangers for nuclear power plants and problems of HTGR

Hiroyuki YAMAO*, Yoshizo OKAMOTO
and Konomo SANOKAWA

Division of Reactor Engineering, Tokai, JAERI

(Received February 22, 1977)

The problems in development of heat exchangers for nuclear reactors have been examined in literature survey through Annual Index Subjects of NSA (Nuclear Science Abstracts) for the past ten years. R and D on heat exchangers for LMFBR, HTGR, LWR and HWR are on the increase. In the case of HTGRs, R and D on heat resisting materials including the corrosion and on hydrogen permeation of heat exchanger walls in high temperature pressure helium environment are important. Future R and D subjects for HTGR heat exchangers in showing the high temperature endurance are presented.

---

*) On leave from Mitsubishi Heavy Industries, Ltd

目　　次

1. 序

多目的高温ガス炉（High-Temperature Gas-cooled Reactor）の開発を推進する上には，高温高圧の条件のもとで，円滑にその機能を発揮し作動する熱交換器を開発することは不可欠の課題である。とくに，冷却材温度が１０００℃という環境では，これまで５００℃程度の温度で使用されていた従来の舶用の原子炉熱交換器には見られない，高温個有の多くの問題に対処することが要求される。

本稿は，原子炉用熱交換器に関して，過去１０年間に，ＮＳＡ（Nuclear Science Abstracts）に掲載された文献の索引を紹介するとともに，それらの文献を用いて熱交換器の開発上の問題点を調査し，その世界的推移，動向を把握することを目的とした。

まず，２，３章では文献の調査方法，および文献の分類区分を説明した。引きつづき，４章ではそれらの結果をもとに，ＨＴＧＲを中心とした各種型式の原子炉用熱交換器開発上の推移，および動向について述べた。また，５章ではとくに，ＨＴＧＲを対象とした熱交換器開発上の問題点を抽出し，その内容について説明を加えた。

2. 文献調査方法

原子炉施設に用いられる熱交換器に関する文献を収集するため，ＮＳＡ〔Vol．21（１９６７）～Vol．33（１９７６）〕のAnnual Index Subjects の中で，Heat excnanger Steam generator 項の索引を利用し文献を抽出した。

これらの文献の索引を，巻末付録Ａに掲載する。索引中には，熱交換器に関する文献を分類する目的で，次のキーワードを抽出して分類を行なった。

(1) 炉型
(2) 熱交換器の型式
(3) 国名
(4) 冷却材

これら４項目に関して，詳しくは表３－１にある記号を用いて分類がされている。

ＮＳＡによる索引番号は，21：17499（Ｐ．Ａ１）のように，索引中に記入されている。本報では，新しい索引番号として，５桁の数字で表わし，前の２桁の数は発行年度を，残りの３桁はその年にＮＳＡに掲載された文献につけた通し番号を示す。１例として，67010 は１９６７年に掲載された文献で　010　は通し番号である。

3. 分類区分

抽出した文献を，(1)炉型，(2)熱交換器の型式，(3)文献発表国，(4)冷却材の種類，および(5)文献の内容別，の５項目に大別し，さらに詳しくそれぞれ表３-１，３-２ のように分類を行なった。

1. 序

多目的高温ガス炉（High-Temperature Gas-cooled Reactor）の開発を推進する上には，高温高圧の条件のもとで，円滑にその機能を発揮し作動する熱交換器を開発することは不可欠の課題である。とくに，冷却材温度が１０００℃という環境では，これまで５００℃程度の温度で使用されていた従来の舶用の原子炉熱交換器には見られない，高温個有の多くの問題に対処することが要求される。

本稿は，原子炉用熱交換器に関して，過去１０年間に，ＮＳＡ（Nuclear Science Abstracts）に掲載された文献の索引を紹介するとともに，それらの文献を用いて熱交換器の開発上の問題点を調査し，その世界的推移，動向を把握することを目的とした。

まず，２，３章では文献の調査方法，および文献の分類区分を説明した。引きつづき，４章ではそれらの結果をもとに，ＨＴＧＲを中心とした各種型式の原子炉用熱交換器開発上の推移，および動向について述べた。また，５章ではとくに，ＨＴＧＲを対象とした熱交換器開発上の問題点を抽出し，その内容について説明を加えた。

2. 文献調査方法

原子炉施設に用いられる熱交換器に関する文献を収集するため，ＮＳＡ〔Vol.21（１９６７）～Vol.33（１９７６）〕のAnnual Index Subjects の中で，Heat excnanger Steam generator 項の索引を利用し文献を抽出した。

これらの文献の索引を，巻末付録Ａに掲載する。索引中には，熱交換器に関する文献を分類する目的で，次のキーワードを抽出して分類を行なった。

(1) 炉型
(2) 熱交換器の型式
(3) 国名
(4) 冷却材

これら４項目に関して，詳しくは表３－１にある記号を用いて分類がされている。

ＮＳＡによる索引番号は，21：17499（Ｐ・Ａ１）のように，索引中に記入されている。本報では，新しい索引番号として，５桁の数字で表わし，前の２桁の数は発行年度を，残りの３桁はその年にＮＳＡに掲載された文献につけた通し番号を示す。１例として，67010 は１９６７年に掲載された文献で　010　は通し番号である。

3. 分類区分

抽出した文献を，(1)炉型，(2)熱交換器の型式，(3)文献発表国，(4)冷却材の種類，および(5)文献の内容別，の５項目に大別し，さらに詳しくそれぞれ表３-１，３-２ のように分類を行なった。

1. 序

多目的高温ガス炉（High-Temperature Gas-cooled Reactor）の開発を推進する上には，高温高圧の条件のもとで，円滑にその機能を発揮し作動する熱交換器を開発することは不可欠の課題である。とくに，冷却材温度が１０００℃という環境では，これまで５００℃程度の温度で使用されていた従来の舶用の原子炉熱交換器には見られない，高温個有の多くの問題に対処することが要求される。

本稿は，原子炉用熱交換器に関して，過去１０年間に，ＮＳＡ（Nuclear Science Abstracts）に掲載された文献の索引を紹介するとともに，それらの文献を用いて熱交換器の開発上の問題点を調査し，その世界的推移，動向を把握することを目的とした。

まず，２，３章では文献の調査方法，および文献の分類区分を説明した。引きつづき，４章ではそれらの結果をもとに，ＨＴＧＲを中心とした各種型式の原子炉用熱交換器開発上の推移，および動向について述べた。また，５章ではとくに，ＨＴＧＲを対象とした熱交換器開発上の問題点を抽出し，その内容について説明を加えた。

2. 文献調査方法

原子炉施設に用いられる熱交換器に関する文献を収集するため，ＮＳＡ［Vol．21（１９６７）～Vol．33（１９７６）］のAnnual Index Subjects の中で，Heat excnanger Steam generator 項の索引を利用し文献を抽出した。

これらの文献の索引を，巻末付録Ａに掲載する。索引中には，熱交換器に関する文献を分類する目的で，次のキーワードを抽出して分類を行なった。

(1) 炉型
(2) 熱交換器の型式
(3) 国名
(4) 冷却材

これら４項目に関して，詳しくは表３－１にある記号を用いて分類がされている。

ＮＳＡによる索引番号は，21：17499（Ｐ・Ａ１）のように，索引中に記入されている。本報では，新しい索引番号として，５桁の数字で表わし，前の２桁の数は発行年度を，残りの３桁はその年にＮＳＡに掲載された文献につけた通し番号を示す。１例として，67010 は１９６７年に掲載された文献で　010　は通し番号である。

3. 分類区分

抽出した文献を，(1)炉型，(2)熱交換器の型式，(3)文献発表国，(4)冷却材の種類，および(5)文献の内容別，の５項目に大別し，さらに詳しくそれぞれ表３-１，３-２ のように分類を行なった。

(1)～(4)は表3-1の記号を用い巻末付録Aの索引中に分類記入し,(5)については文献の内容を，表3-2 のように分類した。(1)～(4)は Heat exchanger,Steam generator の2項目について，(5)ではHeat exchanger 項を対象にして分類してある。Heat exchanger 項の文献を用いて行なった分類の結果を表，および図にまとめた。

表3-3 は，主要5ケ国－アメリカ，フランス，イギリス，西ドイツ，日本を対象とした，各国の年代別の熱交換器に関する発表文献数を示す。

図3-1 は，表3-3 をもとに作成した，各国の年代別の発表文献数の折れ線グラフである。

表3-4 は，冷却材として，Na（LMFBR），He（HTGR，AVR），CO_2（GCR，AGR）のように，冷却材と炉型の関連性を考慮した表を表3-5 に示し，冷却材，炉型を6項に大別して，各国の発表文献数と，その他の文献数を加えた数を，年代別に調べたものである。

図3-2 は，表3-4 をもとに作成した各国の冷却材，炉型別の発表文献数を年代別に，図3-3 は，同じくその総数について，それぞれ棒グラフにしたものである。

表3-6 は，内容別に分類した文献の数を年代別に調べたものであるが，1件の文献でも多種の内容テーマを扱ったものは，それぞれの内容別文献数の1件として，重複して数えてある。

表3-7 は，内容別に分類した文献の索引である。

図3-4 は，表3-6 をもとに作成した，内容別文献数の年代別変化を棒グラフにしたものである。

4. 原子炉用熱交換器の開発の動向

3章で分類し作成した，各国の年代別文献発表数の図3-1 から，各国の熱交換器開発の年代的推移を原子炉の開発と結びつけて分析することは，今後の研究開発の動向を予測し，開発の方向を見定める上で，有効な指針を与えてくれよう。各国の過去10年間の動向を次のようにまとめてみる。

アメリカ　図3-1 に見られる1968，73年の文献数の増加は，図3-2 の冷却材，炉型別文献数の年代的推移を見れば，Na，LMFBR（SEFOR，FFTF）に関係した文献によるものである。LMFBRに関する文献数は他に比べて多いが，近年になってやや減少状態にあり，開発の停滞が推測できる。一方，HTGRについては，1974，75年と発表数が増えつつあり開発の活発化が感じられる。水炉（主に軽水炉）では，過去10年間大きな増加はないものの，毎年1，2の文献が発表されている。これは，その基礎的研究がほぼ終了したことを物語っていよう。

イギリス　文献は10年にわたって，目立った増減なしに発表されているが，1970年は図3-2 よりDragon（HTGR）炉によるもので，73年代はLMFBR（PFR）の開発による。イギリスの初期の開発炉である炭酸ガス冷却炉（GCR）あるいは，その改良型炉（AGR）関係の文献は，1968年以後発表されていない。これはイギリスのAGRからSGHWRへの新型炉開発の移行による状勢変化の影響とみられる。また，73年以後はLMFBR，HTGR共に減少し，原子炉開発の停滞が予想される。

(1)～(4)は表3-1の記号を用い巻末付録Aの索引中に分類記入し，(5)については文献の内容を，表3-2のように分類した。(1)～(4)はHeat exchanger，Steam generatorの2項目について，(5)ではHeat exchanger項を対象にして分類してある。Heat exchanger項の文献を用いて行なった分類の結果を表，および図にまとめた。

表3-3は，主要5ケ国ーアメリカ，フランス，イギリス，西ドイツ，日本を対象とした，各国の年代別の熱交換器に関する発表文献数を示す。

図3-1は，表3-3をもとに作成した，各国の年代別の発表文献数の折れ線グラフである。

表3-4は，冷却材として，Na（LMFBR），He（HTGR，AVR），CO_2（GCR，AGR）のように，冷却材と炉型の関連性を考慮した表を表3-5に示し，冷却材，炉型を6項に大別して，各国の発表文献数と，その他の文献数を加えた数を，年代別に調べたものである。

図3-2は，表3-4をもとに作成した各国の冷却材，炉型別の発表文献数を年代別に，図3-3は，同じくその総数について，それぞれ棒グラフにしたものである。

表3-6は，内容別に分類した文献の数を年代別に調べたものであるが，1件の文献でも多種の内容テーマを扱ったものは，それぞれの内容別文献数の1件として，重複して数えてある。

表3-7は，内容別に分類した文献の索引である。

図3-4は，表3-6をもとに作成した，内容別文献数の年代別変化を棒グラフにしたものである。

4. 原子炉用熱交換器の開発の動向

3章で分類し作成した，各国の年代別文献発表数の図3-1から，各国の熱交換器開発の年代的推移を原子炉の開発と結びつけて分析することは，今後の研究開発の動向を予測し，開発の方向を見定める上で，有効な指針を与えてくれよう。各国の過去10年間の動向を次のようにまとめてみる。

アメリカ　図3-1に見られる1968，73年の文献数の増加は，図3-2の冷却材，炉型別文献数の年代的推移を見れば，Na，LMFBR（SEFOR，FFTF）に関係した文献によるものである。LMFBRに関する文献数は他に比べて多いが，近年になってやや減少状態にあり，開発の停滞が推測できる。一方，HTGRについては，1974，75年と発表数が増えつつあり開発の活発化が感じられる。水炉（主に軽水炉）では，過去10年間大きな増加はないものの，毎年1，2の文献が発表されている。これは，その基礎的研究がほぼ終了したことを物語っていよう。

イギリス　文献は10年にわたって，目立った増減なしに発表されているが，1970年は図3-2よりDragon（HTGR）炉によるもので，73年代はLMFBR（PFR）の開発による。イギリスの初期の開発炉である炭酸ガス冷却炉（GCR）あるいは，その改良型炉（AGR）関係の文献は，1968年以後発表されていない。これはイギリスのAGRからSGHWRへの新型炉開発の移行による状勢変化の影響とみられる。また，73年以後はLMFBR，HTGR共に減少し，原子炉開発の停滞が予想される。

フランス　１９７２～７４年の文献数の増加は，図３-２ からＬＭＦＢＲ（Ｐｈｅｎｉｘ）の開発に負うところが大きい。一方，ＧＣＲの開発によるものとして，ＥＤＦ－１，２，３．ＥＬ－４などの炉に関連した文献が，１９７３年代まで毎年発表されていて，開発炉をＧＣＲに絞っていたフランスの方針がよく表われている。近年のフランスの動向の特徴として，ＧＣＲやＨＴＧＲに比べてＬＭＦＢＲの開発に重点が置かれているといえる。

西ドイツ　１９７３年前後の活発な研究成果の発表は，ＬＭＦＢＲ（ＳＮ－３００），ＰＷＲ，ＨＴＧＲの開発に関連している。

日本　１９７１年の増加はＬＭＦＢＲの影響で，近年になりＨＴＧＲ関係の文献も発表され，全体として，１９６９年以後文献は増加しており，研究開発が進んでいることを示す。

世界的動向　図３-３ の冷却材，炉型別文献の総数の年代変化を見ると，ＬＭＦＢＲについては，文献は他の炉に比べて非常に多く，開発のための多くの研究すべき問題点が伴なっていたことを示す。１９６８年と７３年に文献は増加しているが，７３年以後減少の傾向が見られる。ＨＴＧＲについては，１９７０年のDragon 炉によるものを除けば，近年になって文献は増加を辿っており，活発な開発姿勢を感じる。ＧＣＲ，ＭＨＤ，ＭＳＲに関するものは１９７３年以後文献はない。また，水炉では過去１０年間，ほぼ定常的な発表の傾向にあり，同型炉の建設に関する改良研究として，今後もこの傾向を示すことが考えられる。このように世界的な原子炉開発の趨勢は，近年になってＨＴＧＲ，ＬＭＦＢＲ，水炉（軽および重水炉）に集中されていることから関連して，熱交換器の研究も続けるといってもよく，とくに，ＨＴＧＲではその意欲的な開発状況を示している。

文献内容の動向　過去１０年間に発表された文献の扱っている内容は，炉全体で表３-６ のように，構造，建設的なテーマとして設計，伝熱管，性能，製作，溶接，また，現象的なテーマとして，流れ，熱伝達，振動，および材料と腐食，保守，安全のテーマが多い。過去３年間のテーマの動向としては，図３-４ からわかるように，設計，性能，製作，伝熱管，熱伝達，流れ，振動，応力解析，材料，最適化，腐食，および溶着などのテーマに関する文献が，他の年代に比べて多く発表されており，今後，着手すべき研究テーマの選定の有効な目安となろう。

5. HTGR用熱交換器の開発上の問題点

前章の文献内容のテーマ別の中で，ＨＴＧＲ用熱交換器に限って，その内容上の動向を調べ，問題点をまとめてみた。

本調査では，ＮＳＡ中のHeat exchanger 項だけでなく，Steam generator 項の文献も参考とした。抽出した文献のリストを巻末付録Ｂに掲載した。本リストは，ＨＴＧＲ，ＡＧＲ，およびＭＨＤに関する熱交換器，蒸気発生器についての文献を取りあげており，索引番号と炉型，冷却材，および文献の内容を示すキーワードを記入してある。

表５-１ に内容別に分類した文献数を年代別に示し，それに該当する文献索引を表５-２ に掲載した。

フランス　１９７２～７４年の文献数の増加は，図３-２ からＬＭＦＢＲ（Ｐｈｅｎｉｘ）の開発に負うところが大きい。一方，ＧＣＲの開発によるものとして，ＥＤＦ－１，２，３．ＥＬ－４などの炉に関連した文献が，１９７３年代まで毎年発表されていて，開発炉をＧＣＲに絞っていたフランスの方針がよく表われている。近年のフランスの動向の特徴として，ＧＣＲやＨＴＧＲに比べてＬＭＦＢＲの開発に重点が置かれているといえる。

西ドイツ　１９７３年前後の活発な研究成果の発表は，ＬＭＦＢＲ（ＳＮ－３００），ＰＷＲ，ＨＴＧＲの開発に関連している。

日本　１９７１年の増加はＬＭＦＢＲの影響で，近年になりＨＴＧＲ関係の文献も発表され，全体として，１９６９年以後文献は増加しており，研究開発が進んでいることを示す。

世界的動向　図３-３ の冷却材，炉型別文献の総数の年代変化を見ると，ＬＭＦＢＲについては，文献は他の炉に比べて非常に多く，開発のための多くの研究すべき問題点が伴なっていたことを示す。１９６８年と７３年に文献は増加しているが，７３年以後減少の傾向が見られる。ＨＴＧＲについては，１９７０年のDragon 炉によるものを除けば，近年になって文献は増加を辿っており，活発な開発姿勢を感じる。ＧＣＲ，ＭＨＤ，ＭＳＲに関するものは１９７３年以後文献はない。また，水炉では過去１０年間，ほぼ定常的な発表の傾向にあり，同型炉の建設に関する改良研究として，今後もこの傾向を示すことが考えられる。このように世界的な原子炉開発の趨勢は，近年になってＨＴＧＲ，ＬＭＦＢＲ，水炉（軽および重水炉）に集中されていることから関連して，熱交換器の研究も続けるといってもよく，とくに，ＨＴＧＲではその意欲的な開発状況を示している。

文献内容の動向　過去１０年間に発表された文献の扱っている内容は，炉全体で表３-６ のように，構造，建設的なテーマとして設計，伝熱管，性能，製作，溶接，また，現象的なテーマとして，流れ，熱伝達，振動，および材料と腐食，保守，安全のテーマが多い。過去３年間のテーマの動向としては，図３-４ からわかるように，設計，性能，製作，伝熱管，熱伝達，流れ，振動，応力解析，材料，最適化，腐食，および溶着などのテーマに関する文献が，他の年代に比べて多く発表されており，今後，着手すべき研究テーマの選定の有効な目安となろう。

5. HTGR用熱交換器の開発上の問題点

前章の文献内容のテーマ別の中で，ＨＴＧＲ用熱交換器に限って，その内容上の動向を調べ，問題点をまとめてみた。

本調査では，ＮＳＡ中のHeat exchanger 項だけでなく，Steam generator 項の文献も参考とした。抽出した文献のリストを巻末付録Ｂに掲載した。本リストは，ＨＴＧＲ，ＡＧＲ，およびＭＨＤに関する熱交換器，蒸気発生器についての文献を取りあげており，索引番号と炉型，冷却材，および文献の内容を示すキーワードを記入してある。

表５-１ に内容別に分類した文献数を年代別に示し，それに該当する文献索引を表５-２ に掲載した。

表5-3 は，HTGR用熱交換器の現象的テーマの分類項目の中で，文献数の多い6テーマを選び列記してみた。なお，同表にはすべての文献を炉型の区別なしに内容別に分類した時の総文献数も加え，HTGR用の数と比較検討をし，HTGR用熱交換器の持つ問題点，特徴を調べた。

図5-1 は，表5-3 中の6テーマを扱った文献の総数に対する各テーマの占める比率をグラフにしたもので，HTGR用と炉全体との傾向は似ているが，HTGRでは，流れや熱伝達の問題より材料，腐食の問題が，炉全体の傾向に比べて大きなテーマとなっている。HTGRの特質としての高温環境は，耐熱材料，あるいは腐食の問題を他の炉以上に重要な研究要素としている。反面，熱伝達，流れの問題については，冷却材が単相ガスであるゆえにその流れ，熱伝達の取扱いの容易さは，他の炉に比べこのテーマでの研究を少なくしている，といった理由を考えれば上記の傾向はおのずと理解できる。

表5-4 は，HTGR用熱交換器について研究されているテーマを表5-2 の文献より調査し，キーワードと共に列記したものである。その主項目を列記すると次の様になる。

(1) 冷却材（高温下での物性値の評価，など）
(2) 材料（高温高圧He 中での強度特性，耐食，耐水素透過性金属の選定，セラミック材料，など）
(3) 腐食（耐腐食金属材料の選定，防止対策，など）
(4) 熱伝達（高温高圧He 中での熱伝達，伝熱促進技術，輻射の効果，など）
(5) 振動（耐振設計，流体振動の解析，など）
(6) 流れ（流れの不安定性解析，など）
(7) 水素透過（防止対策，除去方法，など）
(8) ガスタービン（設計上の検討，など）
(9) 高温高圧ループ（実証運転，など）

多目的利用高温ガス炉では１０００℃以上の一次系高温ガスにより，還元ガス，水蒸気を発生させ，製鉄，化学および発電などの多目的利用を意図しているのが特徴であり，従来にない高温度ガスという厳しい環境において使用する熱交換器の設計，開発には多くの解決すべき問題点がある。表5-4 は，過去１０年間に研究されてきたテーマであり，かつ今なお満足すべき答えの出されていない問題をも含んでいよう。そういった問題点を次にまとめてみる。

HTGRのように高温で用いる冷却材としてヘリウム，（Ne，H_2．CO_2）などは，炉内を単相の状態で冷却し，出力密度，および熱流束が幸いにも他の冷却材である水，および液体金属に比べて厳しい条件におかれているわけではないが，高温という条件が，その他の問題を誘発することになる。

冷却材，または構造材の高温下での物性値（比熱，粘性係数，熱伝導率）の変化を正しく評価することは，設計上，基本的な必要不可欠の事項である。

材料については，耐熱，耐食金属材料の多くの調査，開発研究が進められているが，今後は，より実際的な炉内照射試験の確認も必要とされよう。また多目的炉に用いられる水素の伝熱管壁透過は，炉内材料の劣化を招くことになり，その防止法，あるいは除去方法の対策は解決すべき重要な問題である。耐水素透過性材料の選定開発の必要なことはもちろんであるが，材料の強度，あるい

は，耐食性に対する配慮もされた材料の選定，開発の困難さを考慮すると，液体金属介在型の新型熱交換器の開発の必要性も場合により生じる。

腐食対策は，その材料選定と共に，水側ではＬｉＯＨなどの混入による防止，あるいはフレッティング腐食の原因ともなる熱交換器内の流体の流れの不均一，不安定性の防止，あるいは伝熱管の有効な振動防止を意図した設計面の改善にも及ぶ。

伝熱管の振動防止は，腐食防止と関連して，対策を立てる必要があるが，振動は冷却材流体の運動により引起こされる流体振動が支配的であるにもかかわらず，その詳細な機構は充分に解明されていないのが現状であり，現象の調査研究の必要性は大きい。

振動に関連して，熱交換器内の流体の流れは流体振動の起因力として作用する。また，その不均一性により生じる腐食の問題とも関連するばかりでなく，熱伝達特性の良否に大きな影響を与える。

熱伝達は，熱交換器の高熱流束，および高出力密度化を達成する上で必要となる伝熱促進技術の開発という主要な問題を含む。一方，高温高圧ループを使った機器の実証運転も必要とされる。

また，高効率化をねらった直接ガスタービンについての検討も，高温ガス炉の特質を生かす上で重要な課題である。とくに，再生熱交換器の効率はガスタービンプラントの効率を決定する最も主要な課題であり，最適化の研究を進める必要がある。

表５－４ をもとに，ＨＴＧＲ用熱交換器開発上の問題点を明らかにしてきたが，材料と腐食，あるいは腐食と振動と流れ，あるいは流れと熱伝達の関係のように，その因果関係が複雑に結びついており，開発にあたっては，それら個々に秘められた問題を解決するばかりでなく，その結果が他の問題に波及する効果を常に念願においた研究姿勢が望まれよう。

6. あとがき

原子炉施設において使用される熱交換器の開発の動向を，過去１０年間に発表された文献から調査したが，文献調査という大雑把な方法によるにもかかわらず，その文献の発表動向から世界，あるいは各国の開発の推移を辿ることができた。また，ＨＴＧＲ用熱交換器の開発上の問題点を，定量的に把握することはいかぬまでも，その定性的な骨子は浮かび上ってきたと考える。解決すべき問題は，これまで予想しなかった境界領域の問題とも複雑に関連し合っており，解決には，その両立性を考慮した研究姿勢が重要であることを，十分に認識しておく必要があろう。

は，耐食性に対する配慮もされた材料の選定，開発の困難さを考慮すると，液体金属介在型の新型熱交換器の開発の必要性も場合により生じる。

腐食対策は，その材料選定と共に，水側ではＬｉＯＨなどの混入による防止，あるいはフレッティング腐食の原因ともなる熱交換器内の流体の流れの不均一，不安定性の防止，あるいは伝熱管の有効な振動防止を意図した設計面の改善にも及ぶ。

伝熱管の振動防止は，腐食防止と関連して，対策を立てる必要があるが，振動は冷却材流体の運動により引起こされる流体振動が支配的であるにもかかわらず，その詳細な機構は充分に解明されていないのが現状であり，現象の調査研究の必要性は大きい。

振動に関連して，熱交換器内の流体の流れは流体振動の起因力として作用する。また，その不均一性により生じる腐食の問題とも関連するばかりでなく，熱伝達特性の良否に大きな影響を与える。

熱伝達は，熱交換器の高熱流束，および高出力密度化を達成する上で必要となる伝熱促進技術の開発という主要な問題を含む。一方，高温高圧ループを使った機器の実証運転も必要とされる。

また，高効率化をねらった直接ガスタービンについての検討も，高温ガス炉の特質を生かす上で重要な課題である。とくに，再生熱交換器の効率はガスタービンプラントの効率を決定する最も主要な課題であり，最適化の研究を進める必要がある。

表５－４ をもとに，ＨＴＧＲ用熱交換器開発上の問題点を明らかにしてきたが，材料と腐食，あるいは腐食と振動と流れ，あるいは流れと熱伝達の関係のように，その因果関係が複雑に結びついており，開発にあたっては，それら個々に秘められた問題を解決するばかりでなく，その結果が他の問題に波及する効果を常に念願においた研究姿勢が望まれよう。

6. あとがき

原子炉施設において使用される熱交換器の開発の動向を，過去１０年間に発表された文献から調査したが，文献調査という大雑把な方法によるにもかかわらず，その文献の発表動向から世界，あるいは各国の開発の推移を辿ることができた。また，ＨＴＧＲ用熱交換器の開発上の問題点を，定量的に把握することはいかぬまでも，その定性的な骨子は浮かび上ってきたと考える。解決すべき問題は，これまで予想しなかった境界領域の問題とも複雑に関連し合っており，解決には，その両立性を考慮した研究姿勢が重要であることを，十分に認識しておく必要があろう。

表３－１　分類記号一覧表

分　　類	分　類　記　号
炉・型	(1) LWR, PWR, BWR, HWR, PHWR (2) LMFBR, Fast reactor (3) MSRE, MSBR (4) GCR, AGR (5) HTGR, AVR, HTR (6) その他 (MHD, SNAP　)
熱交換器の型式	(1) IHX　(intermediate heat exchanger) (2) SG　(steam generator) (3) RHX　(regenerative heat exchanger) (4) reheater (5) condenser
文献発表国	(1) 米　(U.S.A.)　(5) ソ　(ソ　連) (2) 英　(イギリス)　(6) 日　(日　本) (3) 仏　(フランス)　(7) 伊　(イタリア) (4) 西独　(西ドイツ)　(8) その他
冷　却　材	(1) Na　(6) Gas (2) 液体金属　(7) He (3) 有機材　(8) CO_2 (4) 水　(9) Air (5) 溶融塩　(10) その他

表3－2　内容別分類項目一覧表

	分　　類	記　　号
(1)	熱伝達	(heat transfer)
(2)	質量移動	(mass transfer)
(3)	流れ，対流	(flow, convection)
(4)	応力解析	(stress analysis)
(5)	振動	(vibration)
(6)	過渡現象	(transient phenomena)
(7)	シミュレーション	(simulation)
(8)	デザイン	(design)
(9)	形状，製作，開発	(configuration, fabrication, development)
(10)	性能，仕様	(performance, specification)
(11)	溶接	(welding)
(12)	管群	(tubes, pipes)
(13)	フィン付構造物	(finned elements)
(14)	材料	(material)
(15)	効率	(efficiency)
(16)	価格	(cost)
(17)	最適化	(optimization)
(18)	検査	(inspection)
(19)	保守，清掃	(maintenance, cleaning)
(20)	安全	(safety)
(21)	不具合例	(failures)
(22)	腐食	(corrosion)
(23)	ファウリング	(fouling)
(24)	沈殿	(deposit)

表３－３　各国の熱交換器関係文献数

国＼年度	1967	1968	1969	1970	1971	1972	1973	1974	1975
アメリカ	33	34	21	17	18	18	22	15	10
フランス	8	4	4	7	5	5	7	3	6
イギリス	4	5	2	6	3	10	7	9	2
西ドイツ	4	1	4	6	5	6	13	10	5
日　本				1	3	1	4	3	5
他	8	12	9	13	12	2	7	27	12
総　計	57	56	38	50	46	42	60	67	40

表３－５　冷却材，炉型別分類項目表

区　分　項　目
1. Water, LWR, HWR 注1)
2. Na, Liquid metal, LMFBR
3. He, HTGR, AVR, HTR
4. CO_2, GCR, AGR
5. MSR 注2)
6. MHD

注１）LWRはPWR，BWRを含み，HWRはPHWRを含む

注２）MSRはMSRE，MSBRを含む

表３－４　各国の冷却材，炉型別の文献数の年代別変化

国別	冷却材，炉型	年度								
		1967	1968	1969	1970	1971	1972	1973	1974	1975
アメリカ	Water LWR,HWR	1	1	1		2	1	1		1
	Na, Liquid metal LMFBR	12	15	11	10	2	9	7	6	3
	He HTGR,AVR,HTR								1	3
	CO_2 GCR,AGR									
	MSR	2	4	3	2	3	2			
	MHD		1							
フランス	Water LWR,HWR		1							
	Na, Liquid metal LMFBR	1	2		2	2	1	4	2	
	He HTGR,AVR,HTR							1		
	CO_2 GCR,AGR	1	1	2	2		2	1		
	MSR									
	MHD						1			
イギリス	Water LWR, HWR									
	Na, Liquid metal LMFBR		2	1		3	3	1		
	He HTGR,AVR,HTR		1		7		1	2	1	
	CO_2 GCR, AGR	2	1							
	MSR									
	MHD									
西ドイツ	Water LWR, HWR	1								1
	Na, Liquid metal LMFBR			1	1	2	4	5	1	
	He HTGR,AVR,HTR			1	1	2		2		2
	CO_2 GCR, AGR			1	1		1			
	MSR									
	MHD		1							
日本	Water LWR, HWR									
	Na, Liquid metal LMFBR				1	2		1		1
	He HTGR,AVR,HTR								3	2
	CO_2 GCR, AGR					1				
	MSR									
	MHD									
総数	Water LWR, HWR	2	3	3	2	2	1	1	1	3
	Na, Liquid metal LMFBR	14	21	16	17	14	19	20	17	7
	He HTGR,AVR,HTR	1	1	2	8	2	1	7	5	7
	CO_2 GCR, AGR	3	3	3	3	1	3	2		
	MSR	2	4	3	2	3	2			
	MHD	1	2				1	2		

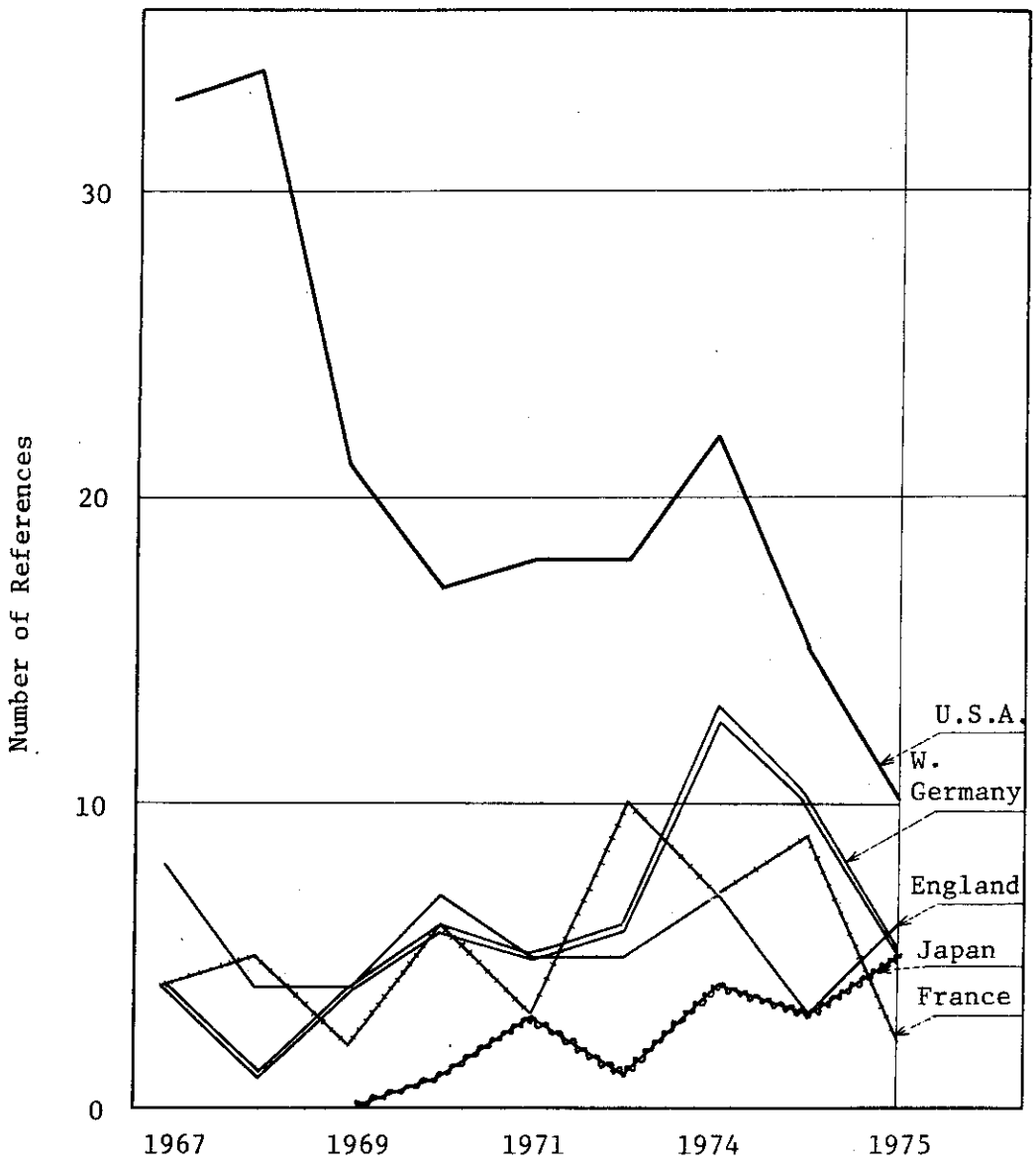

図3－1　各国の熱対換器関係文献発表数

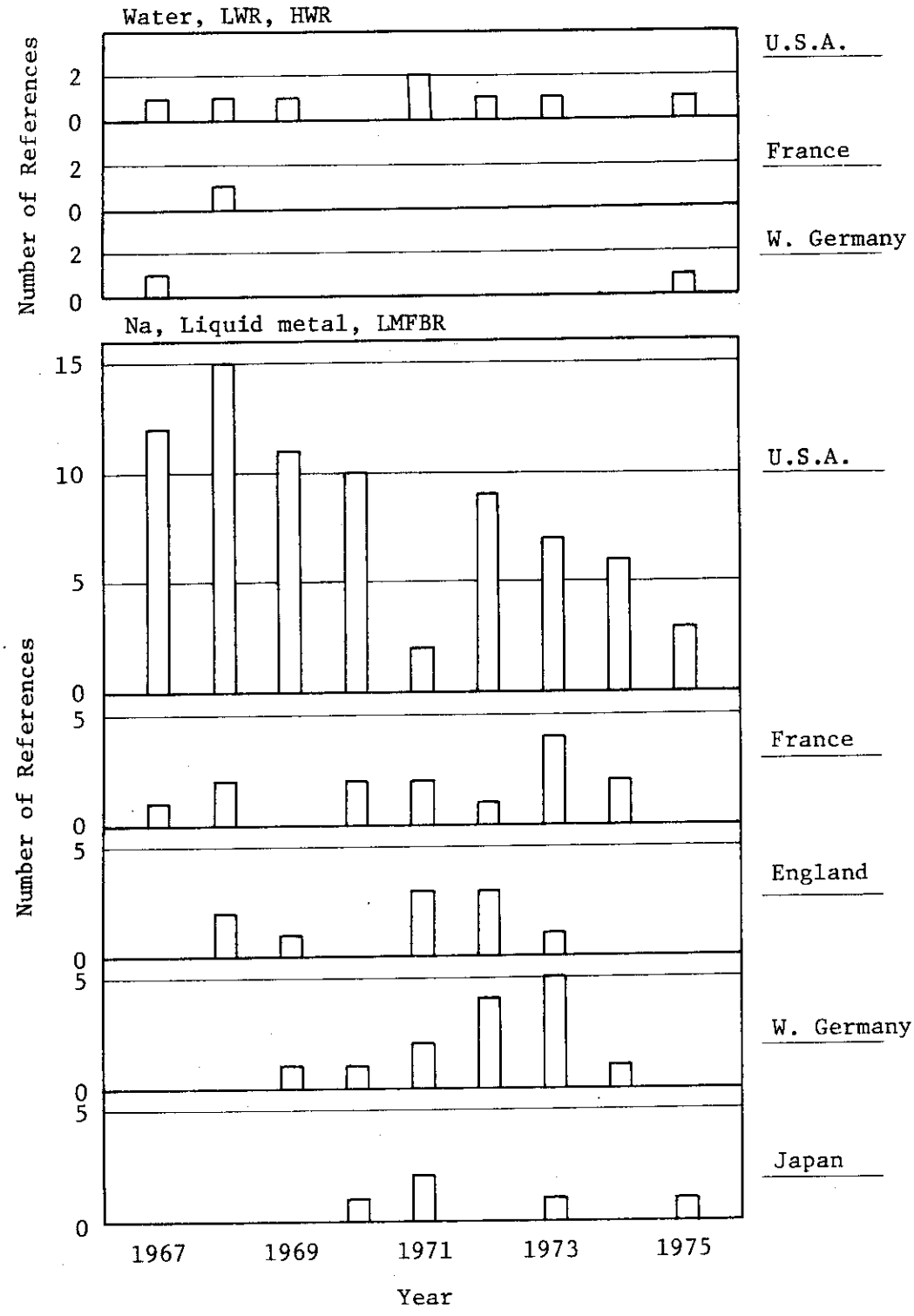

図３－２　各国の冷却材，炉型別の文献数の年代変化

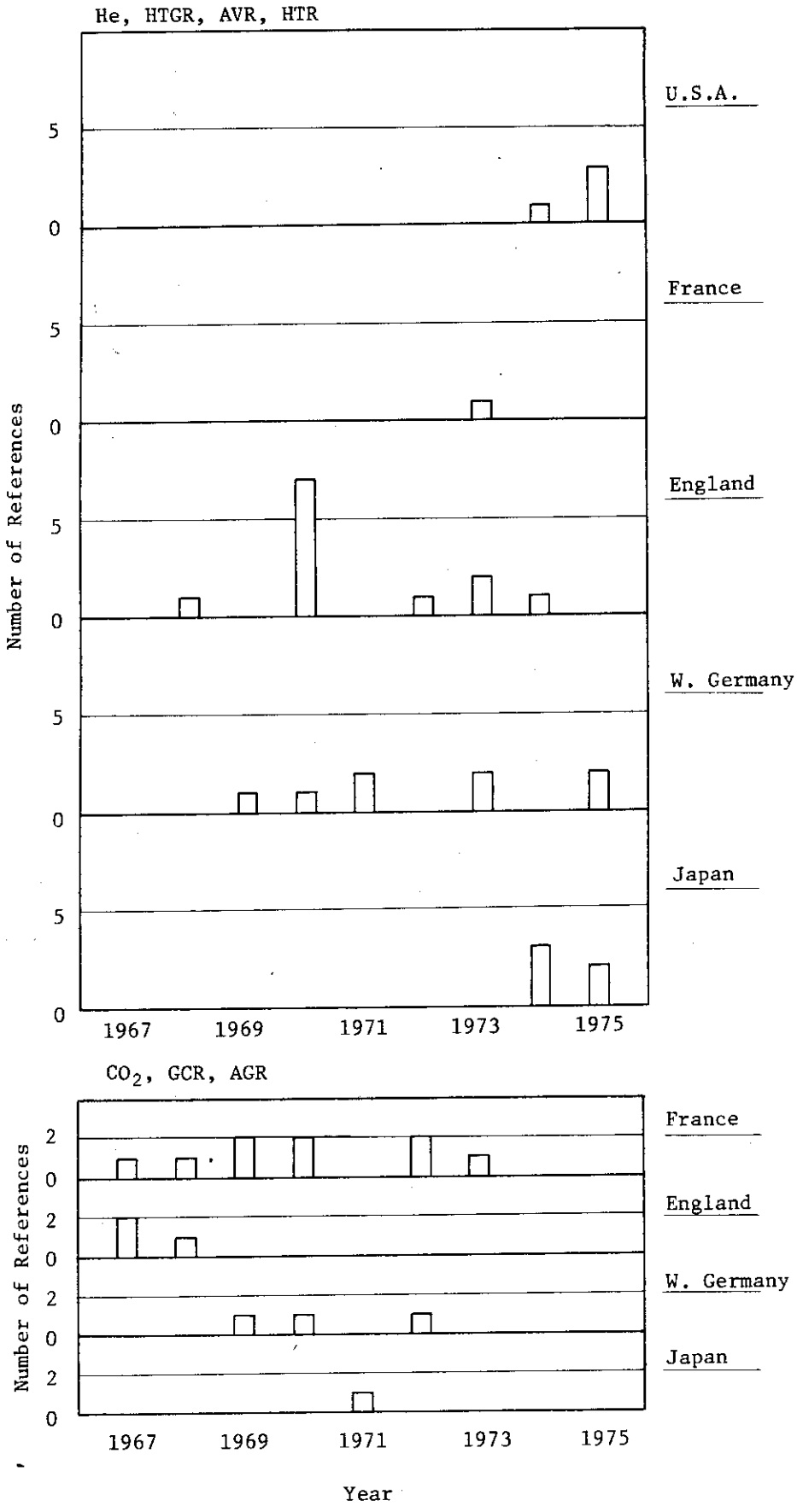
He, HTGR, AVR, HTR
U.S.A.
France
England
W. Germany
Japan
Number of References
5
0
1967
1969
1971
1973
1975
CO_2, GCR, AGR
France
England
W. Germany
Japan
2
Year

図3－2　(続)

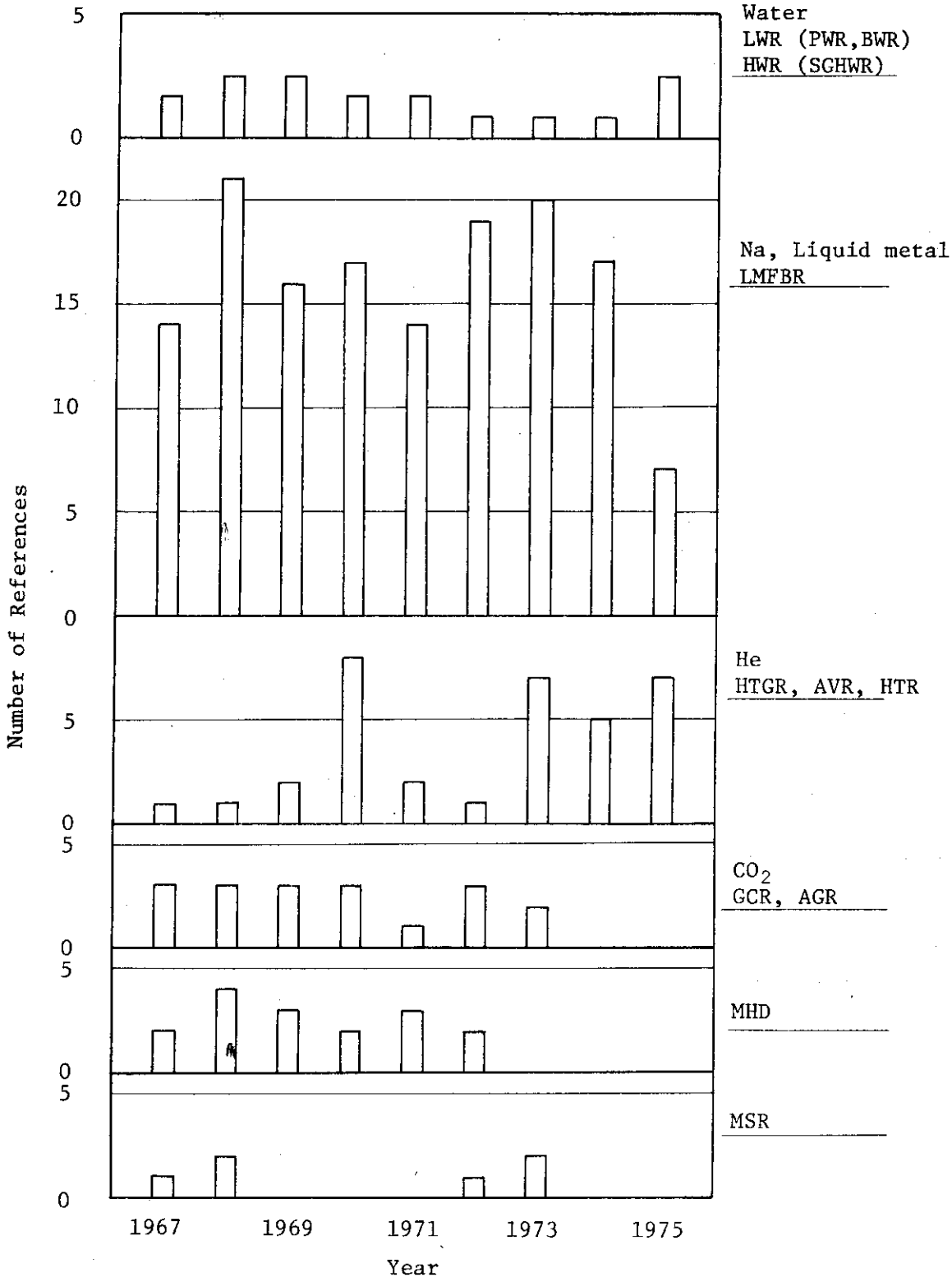

図３－３　冷却材，炉型別文献数の世界的な年代変化

表３－６　内容別文献数の年代別変化

内　　容*	総計 / 年度	1967	1968	1969	1970	1971	1972	1973	1974	1975
Design	120	16	16	10	8	15	14	20	16	5
Tubes, Pipes	91	13	5	15	7	11	10	12	14	4
Flow, Stability	64	3	7	6	6	5	10	6	12	9
Performance	61	10	12	5	5	3	3	3	12	8
Heat transfer	56	11	7	6	6	4	3	4	6	9
Fabrication	30	4	1	3	7	4	1	7	3	
Vibration	29	2	2	4	2	2	5	2	9	1
Material	27	2	3	2	1	3	3	3	6	4
Inspection	24	3	1	5	4	8	1	1	1	
Corrosion	14	2	1		2		2	4	3	
Welding	14	1	1	1	2	3		2	1	3
Optimization	12	1	1	1		2	1	3	3	
Leak	12	2	1	1		4	1	1	1	1
Stress analysis	11	3		1				4	2	1
Deposit Fouling	10	2		1		1	2	3		1
Finned elements	10	2	2	1	1			2	2	
Simulation	7				5					2
Efficiency	6	2	1		1		1			1
Transient phenomena	5	2	1				1			1
Cost	4						3	1		
Mass transfer	2								2	

* Designは Designと Descriptionを含む

Performanceは Performanceと Specificationを含む

Fabricationは Fabricationと Developmentと Configurationを含む

Inspectionは Inspectionと Maintenanceと Safetyと Failuresを含む

表 3－7　内容別文献の索引

contents	index
Design	67005,67006,67007,67008,67010,67011,67014,67016,67019,67026, 67027,67045,67058,67077,67081,67082,68019,68021,68023,68027, 68029,68030,68031,68032,68033,68034,68074,68076,68077,68084, 68087,68088,69002,69004,69005,69012,69014,69020,69021,69033, 69056,69059,70002,70005,70006,70010,70011,70014,70052,70057, 71003,71004,71006,71008,71009,71010,71013,71016,71027,71028, 71030,71044,71049,71050,71052,72006,72009,72010,72011,72013, 72014,72015,72016,72023,72024,72026,72027,72039,72052,73003, 73004,73006,73009,73010,73011,73012,73013,73014,73015,73016, 73017,73018,73019,73020,73021,73052,73054,73055,73056,74011, 74012,74013,74014,74017,74018,74040,74050,74051,74052,74063, 74064,74067,74070,74071,74072,75035,75037,75040,75041,75055
Performance	67049,67050,67051,67052,67053,67054,67055,67056,67057,67080, 68047,68048,68049,68050,68051,68052,68053,68054,68087,68091, 68092,68094,69032,69033,69034,69049,69063,70027,70028,70029, 70030,70061,71023,71024,71027,72040,72041,72042,73003,73030, 73031,74038,74039,74040,74041,74042,74043,74044,74045,74074, 74080,74081,74082,75021,75022,75023,75050,75051,75052,75053, 75054
Fabrication	67001,67030,67034,67083,68008,69022,69023,69026,70018,70019, 70022,70023,70045,70052,70058,71041,71048,71057,71058,72028, 73023,73024,73040,73042,73064,73068,73070,74007,74013,74024
Finned elements	67043,67070,68041,68058,69027,70032,73040,73060,74045,74077
Tubes, Pipes	67019,67031,67032,67038,67040,67042,67044,67046,67063,67064, 67068,67069,67072,68039,68040,68041,68043,68055,69023,69027, 69028,69041,69043,69044,69047,69048,69049,69060,69061,69065, 69066,69067,69068,70014,70032,70036,70044,70045,70040,70048, 71024,71029,71034,71035,71036,71037,71038,71039,71043,71044, 71057,72013,72016,72024,72032,72036,72037,72041,72044,72053, 72054,73010,73011,73033,73034,73036,73037,73038,73040,73041, 73064,73071,73076,74022,74023,74024,74032,74033,74056,74057, 74064,74076,74077,74078,74081,74088,74089,75011,75042,75042, 75047

表３－７　内容別文献の索引（続）

contents	index
Welding	67075,68069,69045,70044,70047,71020,71039,71040,73023,73077,74059,75011,75030,75061
Heat transfer	67036,67037,67038,67039,67040,67041,67042,67043,67044,67045,67084,68003,68009,68039,68040,68041,68042,68043,69027,69028,69029,69030,69060,69061,70024,70049,70050,70052,70053,70054,71018,71019,71053,71057,72036,72037,72055,73010,73019,73026,73062,74017,74038,74074,74075,74076,74077,75014,75015,75016,75023,75034,75042,75043,75044,75053
Flow	67039,67041,67084,68003,68037,68044,68064,68067,68068,68090,69028,69043,69047,69059,69065,69066,70024,70026,70039,70042,70046,70048,71019,71042,71044,71054,71056,72002,72033,72034,72036,72037,72051,72053,72054,72055,72059,73011,73024,73025,73046,73060,73061,74013,74021,74022,74026,74033,74034,74053,74057,74075,74076,74077,74080,75013,75034,75042,75048,75049,75053,75058,75059,75060
Vibration	67068,67069,68067,68068,69043,69044,69065,69066,70046,70048,71038,71043,72053,72054,72057,72058,72059,73038,73042,74030,74031,74032,74033,74034,74053,74057,74058,74089,75019
Stress analysis	67061,67065,67066,69041,73035,73063,73064,73073,74021,74084,75029
Transient phenomena	67065,67066,68052,72042,75015
Simulation Mass transfer	70025,70033,70034,70035,70037,70056 74028,74076
Material	67003,67083,68004,68047,68048,69031,69049,70045,71025,71037,71049,72030,72031,72039,73029,73044,73059,74004,74010,74019,74046,74054,74073,75032,75045,75048,75049
Efficiency	67037,67084,68035,70020,72015,75010
Cost	72030,72031,72044,73047
Optimization	67027,68009,69021,71022,71031,72014,73015,73017,73018,74011,74018,74037

表３－７　内容別文献の索引（続）

contents	index
Inspection	67046,67060,67072,68046,69055,69064,69065,69066,69067,70047,70053,70060,70062,71036,71040,71027,71028,71029,71057,71040,71047,72032,72055,72048
Leak	67047,67079,68078,69068,71020,71039,71055,71056,72038,73037,74027,75018
Corrosion	67003,67004,68048,70003,70055,72002,72003,73004,73037,73046,73062,74010,74023,74062
Deposit, Fouling	67057,67078,69038,71002,72046,72047,73031,73049,73062,75024

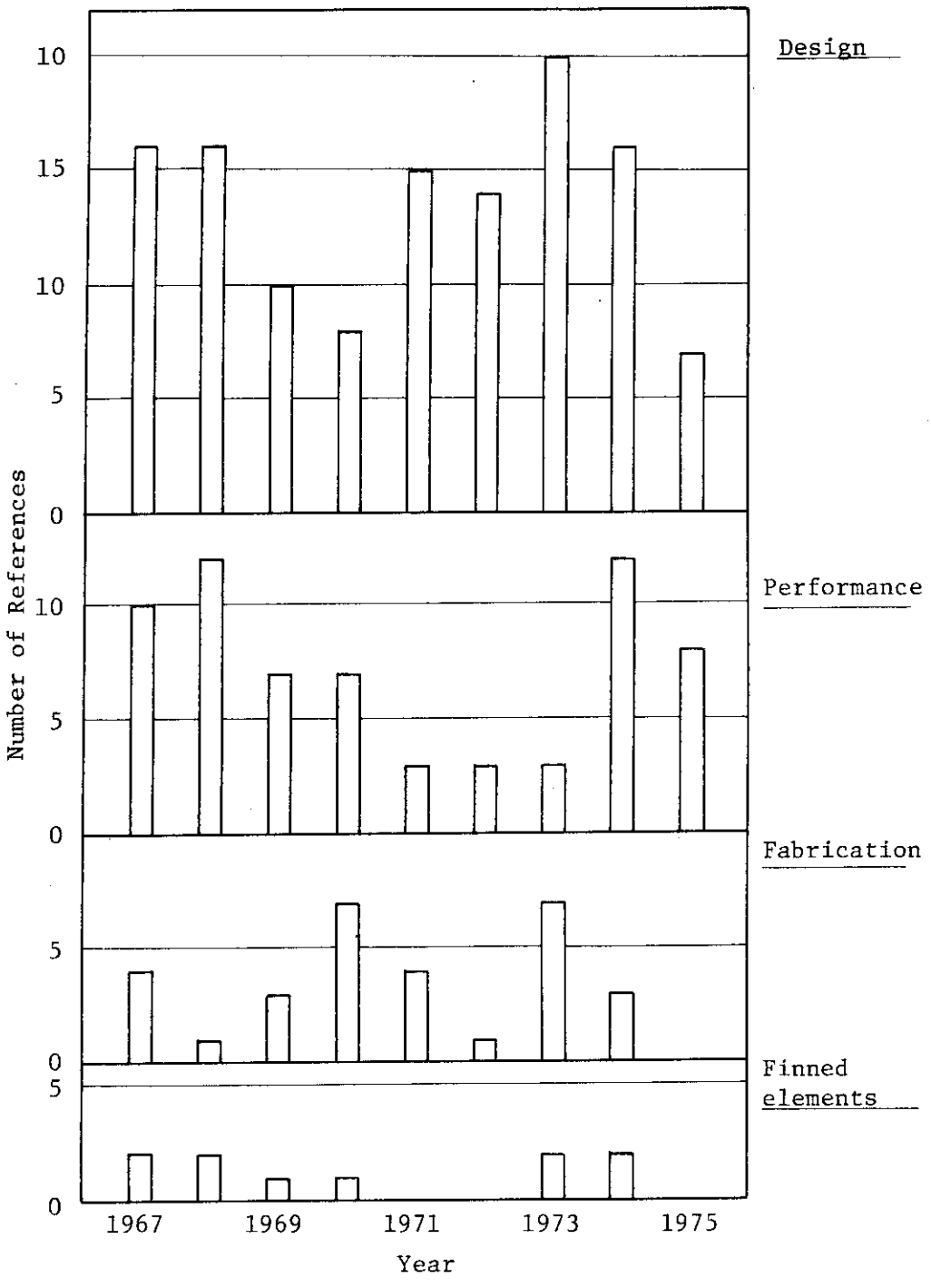

図3－4　内容別文献数の年代別変化

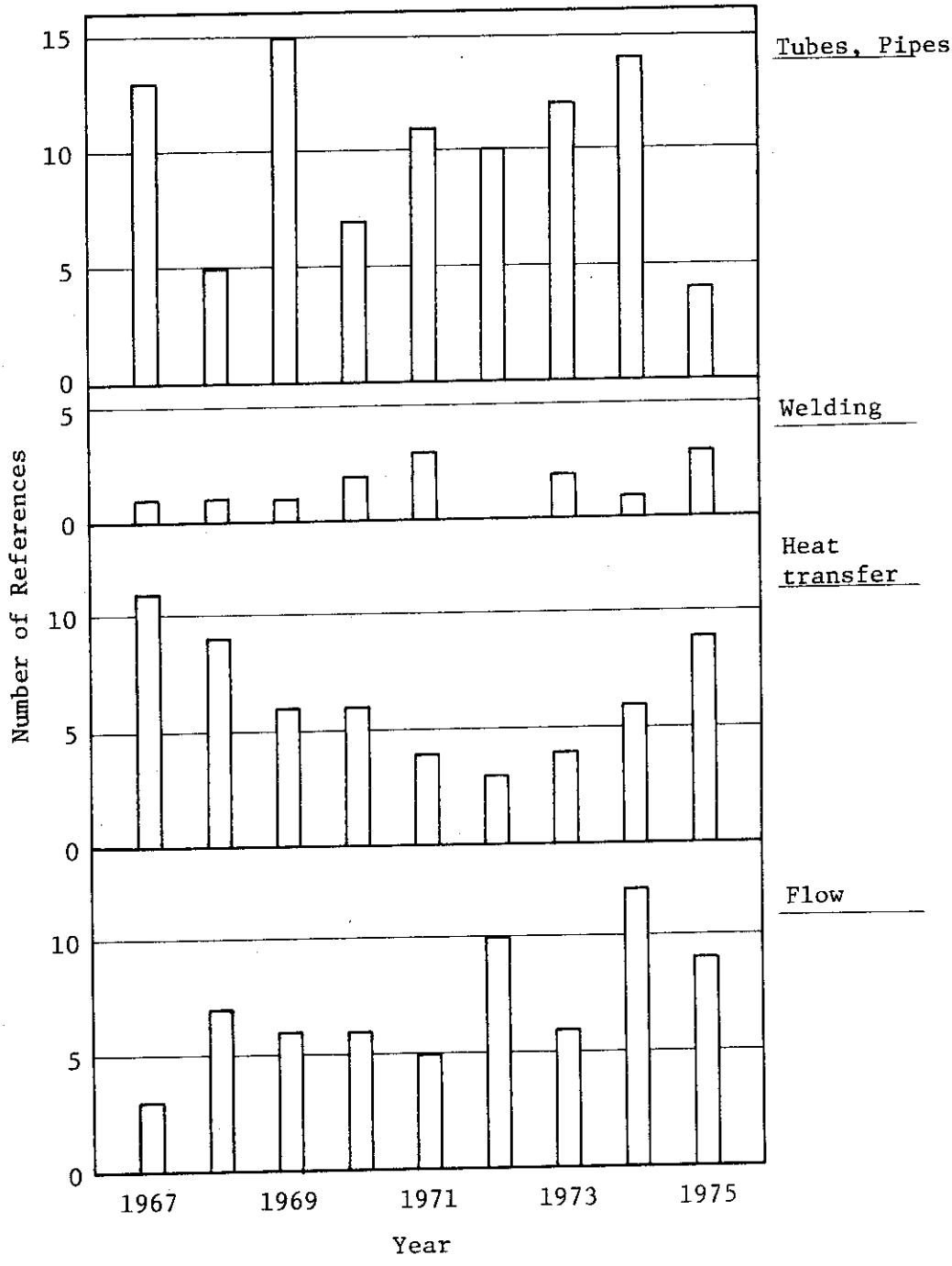
Tubes, Pipes
Welding
Heat transfer
Flow
Number of References
Year
15
10
5
0
1967
1969
1971
1973
1975

図3－4　(続)

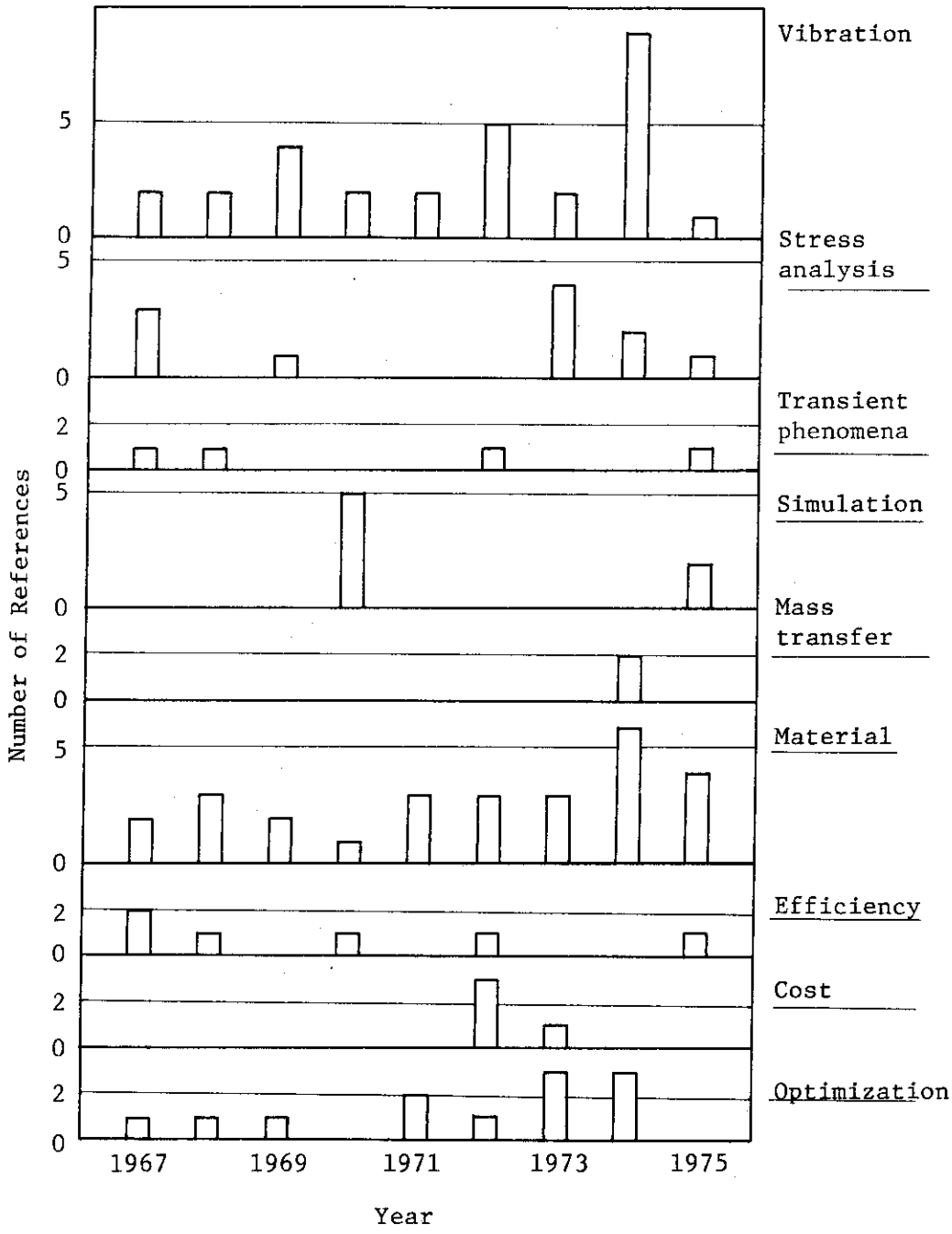
Vibration
Stress analysis
Transient phenomena
Simulation
Mass transfer
Material
Efficiency
Cost
Optimization
Number of References
Year
1967
1969
1971
1973
1975

図3－4　(続)

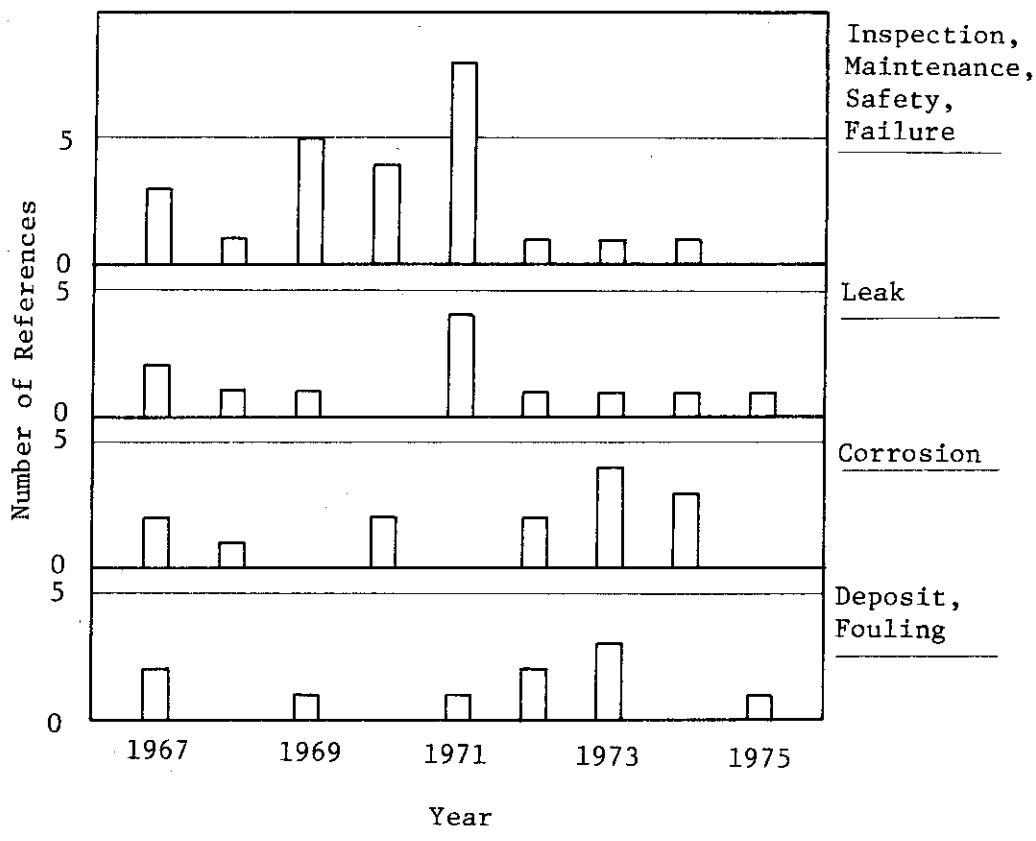
Inspection,
Maintenance,
Safety,
Failure
Leak
Corrosion
Deposit,
Fouling
Number of References
5
0
5
0
5
0
5
0
1967
1969
1971
1973
1975
Year

図3－4　(続)

表5－1　HTGRに関する主要テーマ別文献数

年度 / contents	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	総計
Design	1 (3)		(5)	1 (4)	(4)	(1)	(1)	3 (1)	(1)	(1)	5 (21)
Performance	(1)	(2)		(1)			(1)		2	1	3 (5)
Fabrication		(1)		1	(1)						1 (2)
Development	(1)				(1)						(2)
Material		1		1	(1)		(1)	3 (1)	1 (2)	(1)	6 (6)
Corrosion		1 (1)		1 (1)	(1)	1	1		(1)	(1)	4 (5)
Heat transfer				4	1	1	(1)	(1)	4 (1)	1	11 (3)
Contamination							1				1
Cost							1				
Flow-stability	(3)		(1)	(1)			1	(2)	1 (2)		2 (9)
Flow	(1)		(1)	1		1		(1)	(1)		2 (4)
Vibration	(2)		(2)			(1)	1	(2)	(1)		1 (8)
Welding			(1)	(2)	(1)						(4)
Optimum			1		(1)						1 (1)
Deposit						(2)					(2)
Safety	1			1					(1)		2
Testing	(1)	(1)	(2)	1			(1)				1 (5)
Hydrogen permeation						(1)			2		2 (1)
Gas turbine			1					1			2

(　) 内は Steam generator 項より抽出した文献数

表５－２　HTGR用熱交換器に関する文献の内容別索引表

contents	index
Design	67006,70052,74007,74014,74067,(67003,67022,67036,69010,69031,69025,69043,69074,70009,70015,70027,70024,71022,71025,71019,71023,72029,73009,74013,75093,76017)
Performance	75021,75022,76036,(67067,68079,68080,70094,73078)
Fabrication	70052,(68076,71025)
Development	(67036,71036)
Material	68048,70031,74004,74067,74073,75032,(67075,71087,73009,74114,75005,75084,76007)
Corrosion	68048,70055,72003,73004,(68058,70104,70105,71092,75084,76007)
Heat transfer	70049,70050,70053,70052,71046,72055,75014,75016,75043,75045,76026,(73052,74052,75103)
Contamination	73045
Cost	73047
Flow-stability	73024,75014,(67003,67074,67075,69050,70052,74052,74134,75103,75105)
Flow	70026,72055,(67051,69005,74052,75103)
Vibration	73004,(67074,67075,69083,69084,72093,74034,74035,75039)
Welding	(69098,70104,70105,71092)
Optimum	69021,(71023)
Deposit	(72015,72098)
Safety	67006,70053,(75053)
Testing	70041,(67036,68079,69083,69084,73052)
Hydrogen permeation	75032,75045,(72098)
Gas turbine	69021,74007

(　) 内は Steam generator 項中の索引番号

表５－３　HTGR および炉全体における主要テーマ

テーマ	HTGR	炉全体
	文献総数	文献総数
Flow, Flow-stability	17	64
Heat transfer	14	56
Material	12	27
Vibration	9	29
Corrosion	9	14
Welding	4	14

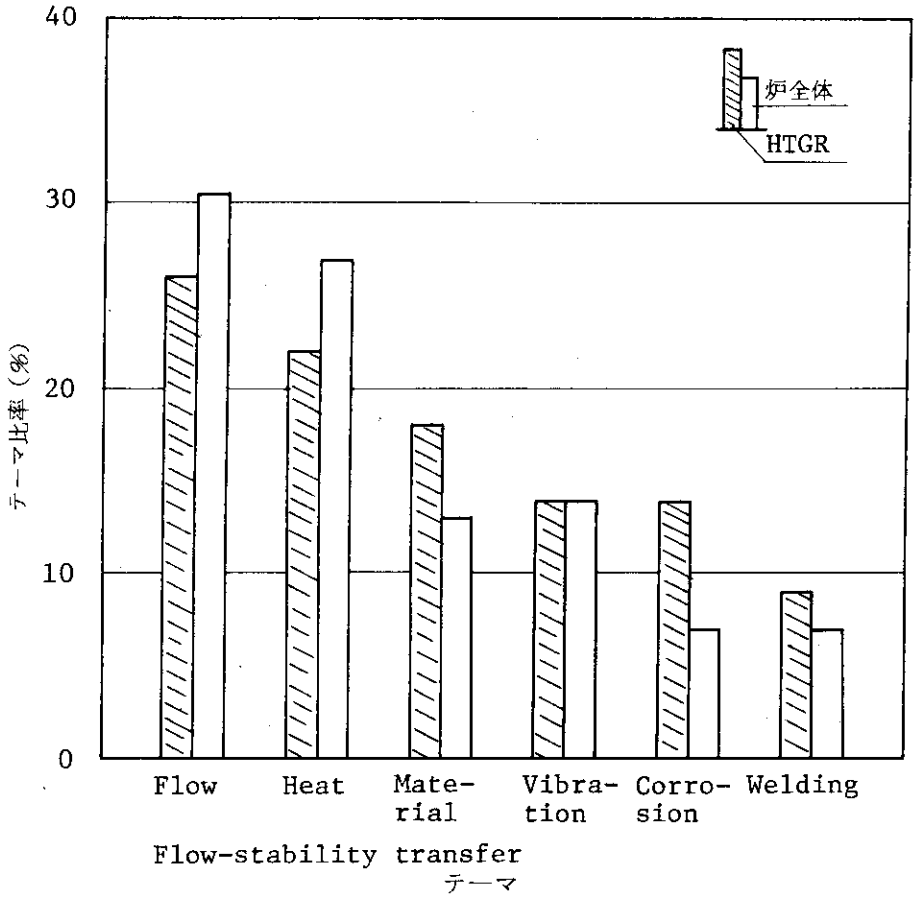

図５－１　HTGR と炉全体における６テーマに対する各テーマの比率と比較

表５－４　HTGR用熱交換器に関する文献で扱われた問題点一覧表

分類	研究テーマ
冷却材	① 高温下での物性値（He，He－空気混合ガス，H_2O） ② 金属材料との適合性
材料	① 伝熱管用材料の高温He中での高温強度特性と耐食性を備えた金属材料の開発 ② 溶接性 ③ 冷却材と金属材料との適合性 ④ セラミック材料の使用 ⑤ 輻射の影響 ⑥ 耐水性透過性金属材料
腐食	① 耐腐食金属の選定．開発 ② 腐食防止対策（LiOHの混入，流れの均一安定化，熱伝達の均一化） ③ 伝熱管の振動によるフレッティング問題（振動の抑制，サポートの改良） ④ 応力腐食による伝熱管の欠損の処置（材料交換）
熱伝達	① 作動媒体の選択（Ne，He，CO_2） ② 高温下での物性値（He，混合ガス） ③ 熱伝達の均一化 ④ 伝熱促進技術（フィン付伝熱面） ⑤ 輻射の効果 ⑥ 高温，高圧 He中でての熱伝達性能試験 ⑦ ヘリカルコイルチューブ式熱交換器の性能試験 ⑧ 蒸気発生器の膜沸騰熱伝達時の安全マージン評価 ⑨ 伝熱面の縮小化
振動	① 耐震設計 ② 数学的振動モデル（空力，音響，地震による起因力を考慮）を用いた解析とモデル試験 ③ 流体振動によるフレッティング問題，防止策 ④ 流れの安定性，均一性を企るデザイン

分類	研究テーマ
流れ	① 流れの不安定性解析，モデル試験 ② 流体振動抑制のための流れの均一化を企ったデザイン ③ 流れの速度，温度分布測定
溶接	① 溶接性
溶着	① 高温化の侵炭問題
安全	① 製作，建設時の安全要求 ② 保守と検査 ③ 遠隔操作機器の開発 ④ 膜沸騰時の安全マージン評価 ⑤ 伝熱管の寿命評価 ⑥ 過渡現象の予測計算 ⑦ 伝熱管プラギング装置
水素透過	① 水素透過特性試験 ② 水素除去方法（チタンスポンジ） ③ 水素透過防止方法（コーティング） ④ 新型熱交換器（液体金属介在型）
汚染	① 二次系へのＦＰ混入による汚染評価
ガスタービン	① 直接ガスタービンクローズドサイクルの設計．検討 ② 作動流体の選定（He，Ne，CO_2）
高温高圧ループ	① 実証運転（高温機器の性能，信頼性）
設計	① 蒸気発生器のプラント内配置問題 ② ヘリカルコイルチューブ ③ 熱伝達，流れの均一化 ④ 伝熱面の縮小

JAERI-M 7034

付 録 A

Heat exchangers 1967 Vol. 21

67000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators				
1	construction and control methods for graphite-gas,review of, 21:17499	仏		CO_2	Marcoule Chinon(GCR)
2	coolant-duct seals for, design of, 21:8007(P)	英			
3	corrosion of metallic, effects of heat transfer phenomena on, 21:39553	英			
4	corrosion of sodium--water, 21:30972			Na	
5	design and fabrication for Delft Molten Salt Project, 21:43087 (DMSP-G-8)	オランダ			
6	design and performance of AVR, 21:12173	西独		He	AVR
7	design for 250 MW(e) heavy water reactor, 21:4650	西独		重水	HWR
8	design for dissipating accelerator heat, engineering drawings for, 21:6122	米			
9	design for gas-cooled reactors, 21:46909(P)				GCR
10	design for liquid sodium loop for operation at 800°C, 21:30969	オーストリア		Na	
11	design for organic cooled reactors, 21:21709	伊		有機材	
12	design for power reactors, 21:31992(P)				
13	design for steam generators using liquid metals as heating media, 21:36257(P)			液体金属	
14	design for studying heat-transfer properties of liquid metals, 21:41588	仏	IHX SG	液体金属	
15	design for use with reactors, 21:23776(P)				
16	design of bellows-free piping system for, for sodium Components Test Installation, 21:6128 (NAA-SR-Memo-12018)	米		Na	
17	design of circular cross-section, 21:1498(P)	英			
18	design of coolers for use with reactor loading machines, 21:32045(P)				
19	design of double-pipe, mathematical analysis for, 21:43146	米			
20	design of improved, for nuclear power plants, 21:43127(P)				
21	design of improved, to transfer heat from hot gaseous fluids, 21:8525(P)	米			
22	design of liquid metal, using double tube arrangement and intermediate heat transfer fluid, 21:17763(P)				
23	design of movable, for gas-cooled reactors, 21:5668(P)				GCR
24	design of sheath duct, for sodium or other liquid metals, 21:39101(P)			Na, 液体金属	
25	design of shielded, for gas cooled reactors, 21:40648(P)				GCR
26	design of silver and lithium vapor, for use in space reactor, engineering drawings for, 21:10526	米			スペース
27	design of SNAP-8, optimization of, 21:44916 (N-67-22845)	米			SNAP-8
28	design of stainless steel-lined zirconium alloy, for pressurized-water reactors, 21:46976(P)				PWR
29	design of vortex flow, 21:39109(P)				
30	development of cryogenic fluid, for nuclear rocket propulsion systems, 21:43131 (LA-DC-8716)	米			

31	development of finned tubes for, for increased thermal exchange surface, 21:10520(P)			
32	dynamics of double tube, 21:28156 (SGAE-RT-9/1967)	西独		
32	effect of velocity field on dynamic behavior of double tube, 21:28156 (SGAE-RT-9/1967)	西独		
33	equipment for supporting, design of, 21:31987(P)			
34	fabrication of, for potassium cooling systems, 21:4919 (ORNL-TM-1600)	米	K	
35	finned tubes for, design of tapered, 21:19829(P)			
36	heat transfer analysis of MSRE, 21:35627	米		MSRE
37	heat transfer efficiency of, effects of heating wall calefaction spots on, 21:41138	仏		
38	heat transfer in double-pipe, liquid metal, 21:36781 (BNL-50054)	米	液体金属	
39	heat transfer in longitudinal laminar flow in, effect of flow rate on, 21:229	チェコ		
40	heat transfer in tube,(T), 21:43098(T) (AEC-tr-6893)	米		
41	heat transfer in two-phase flow in, use of differential equations for analysis of, 21:30610 (EUR-3459.e)	コーラトム		
42	heat transfer measurements in double-pipe, 21:31898(R) (ANL-7329)	米		
43	heat transfer to finned surfaces in, 21:4665	日		
44	heat transfer to water in 180°-bent tube bundles of, 21:43144	英		
45	heat-transfer rates in, for liquid-metal systems, design improvements in, 21:30624	米	液体金属	
46	inspection of thin-walled tubes for, use of eddy currents, infrared radiation, and ultrasonics for, 21:10557 (BNWL-SA-404)	米		
47	leakage control methods in dual-purpose boiling water reactors, 21:21750 (ACNP-65568)	米	水	BWR
48	mounting method for, in pressure vessels, 21:2863(P)			
49	performance evaluation of, use of Sodium Components Test Installation for, 21:35590(R) (UNC-5182)	米	Na	
50	performance evaluation program for sodium-cooled reactors, 21:2830(R) (UNC-5162)	米	IHX,SG Na	
51	performance of cryogenic-fluid, using Teflon-lined tubes, 21:41098 (LA-DC-8659)	米		
52	performance of liquid sodium, 21:23622(R) (UNC-5166)	米	IHX,SG Na	
53	performance of MOSEL, calculations of, 21:27685 (JUL-422-RG)	西独	IHX,SG	
54	performance of sodium-cooled, for reactors, 21:25637(R) (UNC-5174)	米	IHX,SG Na	
55	performance of, conference on, 21:41118	米		
56	performance of, effects of exchange surface roughness on, 21:43136	ソ		
57	performance of, effects of surface fouling on, 21:41116	英		
58	pressure-tube support design for reactor, 21:46978(P)			
59	review of 1966 literature on, 21:41119	英		
60	safety device design for system of, 21:23677(P)			
61	steel plates for, determination of stresses in, 21:32868(T) (WAPD-Trans-54)	仏		

62	testing for quality of Hanford Reactor, method for nondestructive, 21:36296 (BNWL-SA-901)	米			Hanford炉
63	testing of double-pipe, 21:23589(R) (ANL-7308)	米			
64	testing Teflon tubes for cryogenic, 21:41098 (LA-DC-8659)	米			
65	transient dynamics and thermal stress analysis of, for nuclear rocket, 21:12574 (N-66-39722)	米			ロケット
66	transient thermal stress analysis of, method for, 21:6129 (NAA-SR-Memo-12129)	米	IHX,SG		
67	tube bundles for, design of support system for, 21:43114(P)				
68	tube vibration and repair in HFBR, 21:35732	米			HFBR
69	tube vibrations in HFBR, repair of, 21:38658 (BNL-11585)	米			HFBR
70	tubes for, design of finned, 21:19512(P)				
71	tubes for, design of grooved-profile, 21:36255(P)				
72	tubing for, ultrasonic inspection of tapered, 21:245	米			
73	use in isotopic heated thermoelectric generator, 21:3223(P)				
74	use of radioisotopic, for attitude control engine, 21:8551(P)				
75	welding methods for, 21:4920	英			GCR
76	HEAT EXCHANGERS (GAS-GAS) mass exchange in low-temperature MHD, 21:9362	ソ			MHD
77	HEAT EXCHANGERS (GAS-LIQUID) design and fabrication of, for high pressure system, 21:41115	英			
78	fouling in dust suspension cooling systems, use of graphite powder in carbon dioxide gas in studies of, 21:17772 (AEEW-R-509)	英		CO_2	
79	leak protection system for reactor, 21:1183(T) (ORNL-tr-1309)				
80	performance testing methods for, 21:41117	英			
81	HEAT EXCHANGERS (LIQUID-LIQUID) design for 1000 MW(e) fused-salt breeder reactor, 21:40570 (ORNL-TM-1545)	米			MSBR
82	design of 30-MW sodium, for reactor cooling system, 21:15640 (APAE-112(Vol.1))	米	IHX,SG	Na	
83	development for metals, 21:30093(R) (ANL-7317)	米			
84	heat transfer efficiency in, with countercurrent turbulent liquid-metal flow, 21:40694 (ANL-7349, pp 92-109)	米		液体金属	

1968 Vol. 22

68000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators				
1	afterheat generation in MSBR fuel loop, 22:7816(R) (ORNL-TM-2057)	米			MSBR
2	analysis of high temperature liquid carbon--niobium--uranium alloy, for high pressure hydrogen, 22:49022	米			
	analysis of high temperature liquid carbon--uranium--zirconium alloy, for high pressure hydrogen, 22:49022	米			
3	annular geometry in, heat transfer for fully developed turbulent flow in concentric, 22:16765	米			
4	brazing alloys for shell-and-tube type, review of, 22:45758	カ			
5	cell structure of, effects on burnout of, 22:227(P)	米			
6	cladding process for, arc strip, 22:15164				
7	configuration of power reactor multiple hairpin-loop pressure vessel-enclosed, description of, 22:48706(P)	英			
8	construction for gas-cooled graphite-moderated power reactors, standards for steel, 22:30183				GCR
9	control by feedback optimization in, description of outlet coolant temperature dynamics system for, 22:20948				
10	control valves in large water-to-hydrogen, interactions with oscillations, 22:45241 (LA-DC-9781)	米		H_2-H_2O	
11	description and arrangement of reactor, 22:18611(P)	米			
12	description for gas-cooled reactors, 22:18635(P)				GCR
13	description of auxiliary multiple, for fission product removal from liquid-cooled liquid-moderated reactors, 22:16342(P)			液体	
14	description of compact two-pass, for nuclear reactor, 22:46974(P)				
15	description of counter current, with variable flow rate, 22:3953(P)				
16	description of gas-cooled reactor prefabricated removable, 22:20885(P)				GCR
17	description of in-reactor-pressure-vessel, 22:20963(P)				
18	description of sealed, for irradiation facility, 22:3956(P)				
19	design for glove box evaporator, 22:14644	米	evaporator		
20	design for MHD generators, 22:15393(P)				MHD
21	design for shipboard water desalination, 22:10161(T) (AEC-tr-6909)	ソ	evaporator		
22	design for sodium-cooled reactors, 22:12506(P)			Na	
23	design of calciner off gas, for nitric acid recovery system, engineering drawings for, 22:25589	米			
24	design of double-tube, 22:42968(P)				
25	design of improved, using fins in fluid path, 22:42969(P)				
26	design of radiation-type, 22:50072(P)				
27	design of regenerative, 22:6533	ソ	RHX		
28	design of shell-and-tube, for gas-cooled reactors, 22:1908(P)				GCR

29	design parameters for sodium cooled fast breeder reactor, 22:16205	米	IHX SG	Na	LMFBR
30	design, fabrication, and operation of, to heat mixtures of cesium and helium to 2000°K, 22:37923	伊			
31	development for MHD thermoelectric power plants, 22:33592	米			MHD
32	development of liquid metal-to-air, for MCR primary loop, 22:32927 (GMAD-3078-33)	米		液体金属	
33	development of sodium to steam, 22:8247(R) (BAW-1280-47)	米	SG	Na-水	
34	development of sodium to steam, 22:16735(R) (BAW-1280-48)	米	SG	Na-水	
35	efficiency of liquid metal nonsymmetrical-duct, 22:30118 (ANL-7438)	米		Na	
36	finned tubes on, machine design for automatic milling of, 22:27984(P)				
37	fluid flow in, control of organic coolant, 22:3868	伊		有機材	
38	fluid flow through, controlling temperature of fluid in, 22:4351(P)				
37	heat transfer by liquid metals in double-pipe, 22:647(R) (BNL-50077)	米		液体金属	
40	heat transfer from in-bed horizontal multiple-tube, in radioactive waste processing plant, 22:37854	米			
41	heat transfer from longitudinally finned tubes of, method of calculating, 22:2301	チェコ			
42	heat transfer in countercurrent annular potassium--sodium, 22:12522	仏		NaK	
43	heat transfer in double-pipe, from liquid metals, 22:28003	米		液体金属	
44	kinetics of cross flow, mathematical model for, 22:28000	英			
45	liquid metal-cooled tube-and-shell, for fast breeder pebble-bed reactors, 22:35490(P)			液体金属	FBR
46	maintenance of EDF-1 Reactor, 22:37452	仏		CO_2	EDF-1(GCR)
47	materials evaluation for AGR, 22:20830	英			AGR
48	materials for AGR-type reactor, requirements for corrosion resistance of, 22:47660 (CONF-671007-2)	英			HTGR,AGR
46	operation of EDF-1 Reactor, 22:37452	仏		CO_2	EDF-1(GCR)
47	performance in sodium-cooled reactors, 22:647(R) (BNL-50077)	米		Na	
48	performance in sodium-cooled reactors, evaluation of thermal-hydraulic and structural, (E), 22:32788(R) (UNC-5213)	米		Na	
49	performance in waste calcining facility, 22:4231(R) (IN-1087)	米			
50	performance of intermediate sodium, 22:48982(R) (LMEC-68-20)	米	IHX SG	Na	
51	performance of liquid-metal, 22:10497	ソ		液体金属	
52	performance of sodium-cooled, under transient operating conditions, 22:44840(R) (UNC-5221)	米	IHX	Na	
53	performance of, for ^{3}He--^{4}He refrigeration system, (T), 22:50093 (JINR-P13-4014)	ソ			
54	performance testing of, for sodium-cooled reactors, 22:10441(R) (UNC-5189)	米	IHX SG	Na	
55	pipes in, surface cleaning and passivation of, 22:12483	仏			
56	size of steam-water, computer calculation of optimum, 22:42948	ユーゴ		水ー蒸気	

57	supports for reactor, design of coolant submerged, 22:1896(P)			
58	testing of finned, for use in evaluation of VAPOTRON principle for boiling water reactors, 22:1759 (EURAEC-1890)	仏	水	BWR
59	tube assemblies for, design of support system for, 22:12514(P)			
60	tube-to-sheet welded joint design for, 22:226(P)			
61	tubes for, design of device for changing, 22:30765(P)			
62	tubes for, design of machine tool for fabrication of finned, 22:10480(P)			
63	tubular element cell design for, 22:227(P)			
64	two-phase cocurrent flow in, review of applications of, 22:12530	伊		
65	use for fission product removal from gas-cooled reactor coolant, description of, 22:37497(P)			GCR
66	use in nuclear MHD power plants, engineering problems in, 22:37503	西独		MHD
67	vibration in, effects of fluid flow rate on, (T), 22:33176	米		
68	vibrations in HFBR, analysis of coolant-flow induced, 22:24991	米		HFBR
69	welding methods for high-temperature, 22:12991	米		
70	HEAT EXCHANGERS (GAS-GAS) description of pressurized carbon dioxide-supplied, for gas-cooled reactors, 22:10042(P)		CO_2	GCR
71	HEAT EXCHANGERS (GAS-LIQUID) configuration of helically coiled tubes in telescoping multi-section concentric cylinder, description of, 22:37496(P)			
72	configuration of integral gas-cooled reactor core-enclosing annular-arranged multiple-tube, description of upward-core-flow, 22:53312(P)			GCR
73	configuration of integral gas-cooled reactor vertically-stacked multiple horizontal flat tube oil, description of downward-core-flow, 22:53311(P)			GCR
74	description of desalination reactor steam-brine heater, 22:30289 (ORNL-TM-2049)	米 SG		
75	description of dual liquid-carbon dioxide, for gas-cooled reactors, 22:10040(P)		CO_2	GCR
76	design of 500-kW, for heat transfer from liquid potassium--sodium to air, 22:10484 (ORNL-4159)	米	NaK-air	
77	design of, computer program for, 22:53497 (K-1750)	米		
78	leak-detection system for power reactor, description of, 22:35519(P)			
74	operation of desalination reactor steam-brine heater, effects of brine velocity and pressure on, 22:30289 (ORNL-TM-2049)	米 SG		
80	tube bank multiple self-supporting ladder-type side-to-side flow, description of gas-cooled reactor, 22:39799(P)			GCR
81	HEAT EXCHANGERS (LIQUID-LIQUID) configuration of liquid metal-cooled reactor forced-circulation closed-circuit dual-flow, description of, 22:50772(P)		液体金属	

82	configuration of liquid metal-cooled reactor, with double-wall multiple ladder-shaped back-and-forth flow tube panels, 22:53324(P)			液体金属	
83	configuration of multiple bundle end-supperted six parallel layer-arranged central tube-connected, 22:39917(P)				
84	description of BN-350 Reactor intermediate, 22:18506	ソ	IHX	Na	BN-350(FBR)
85	description of closed-path baffle-control flow in dual-tube-bank, 22:25178(P)				
86	description of variable length, with fluid tight joints for extensions, 22:1911(P)				
87	design and performance data for FTR, 22:14250(R) (BNWL-470)	米			FTR
88	design of MSRE primary, 22:20819 (ORNL-TM-2098)	米			MSRE
89	effects of liquid sodium environment on fast reactor, 22:18503	仏	IHX,SG	Na	
90	flow in 1000 MW(e) LMFBR, description of liquid sodium, 22:9990(R) (BAW-1312)	米		Na	LMFBR
91	performance history of MSRE primary, 22:20819 (ORNL-TM-2098)	米			MSRE
92	performance in sodium-cooled reactors, evaluation of thermal-hydraulic and structural, 22:20823(R) (UNC-5200)	米	IHX SG	Na	
93	radiation monitoring of MSRE, gamma, 22:12035(R) (ORNL-4191)	米			MSRE
94	thermal performance of Fermi Fast Breeder Reactor intermediate, analysis of, 22:32843	米	IHX	Na	Fermi(FBR)

1969 Vol. 23

69000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators				
1	analysis of MSRE, for fission products by lithium-activated germanium γ spectrometry, 23:9306 (ORNL-4343, pp 62-84)	米			MSRE
2	description of EDF-4 Reactor, 23:1989	仏		CO_2	EDF-4(GCR)
3	description of liquid metal-cooled reactor multiple plate-type concave and convex dimple-arranged parallel-flow, 23:2024(P)			液体金属	
4	design and testing for SCTI, 23:9609(R) (LMEC-68-1)	米		Na	LMFBR
5	design characteristics of sodium-heated intermediate, state-of-the-art survey of, 23:33078 (WARD-2000-23)	米	IHX	Na	LMFBR
6	design for easy installation and removal from nuclear reactors, 23:21865(P)				
7	design for gas-cooled reactors, 23:33300(P)				GCR
8	design for liquid metal-cooled reactors, 23:33301(P)				
9	design for nuclear power plants, 23:21866(P)				
10	design for use with heat carrying ducts for temperature control at remote locations, 23:50013(P)				
11	design of baffled, for nuclear power plants, 23:17671(P)				
12	design of double-walled, for hazardous service, engineering drawings for, 23:2469 (CAPE-1784)	米			
13	design of double-walled, for improved heat transfer, 23:29055(P)				
14	design of implantable, for human blood, 23:29096 (WANL-3857-1)	米			
15	design of improved liquid metal, 23:16025(P)				
16	design of recuperative tubular ceramic, 23:9619(P)				
17	design of tubular, 23:29042(P)				
18	design of tubular, for marine reactors, 23:40937(P)				マリン
19	design of tubular, for nuclear power plants, 23:31460(P)				
20	design of zirconium, for use with sulfuric acid, 23:46005	西独			
21	design parameters for gas cooled reactor, economics and optimization of, 23:28369	西独		He,Ne,CO_2	HTR,AGR GCR
22	development for sodium, 23:35980 (BNL-50170)	米		Na	
23	development of helical coiled-and straight tube-bundle, for sodium-cooled fast nuclear power plants, 23:47227 (ANL-7520(Pt.1), pp 515-26)	伊	IHX,SG	Na	LMFBR
24	dynamics of Benson type, for ORGEL Reactor, (E/T), 23:33035 (EUR-4254)				Orgel
25	elements for, design of curved plate, 23:33302(P)				
26	fabrication of Dodewaard Power Reactor, during 1967, 23:19397(T) (AEC-tr-7032)	オランダ		水	Dodewaard(BWR)
27	finned elements in, heat transfer and pressure drop in batch-produced, 23:21887	英			
28	heat transfer and pressure drop in bundle-tube, at high Reynolds numbers, effects of tube row number on, (E), 23:4774	西独			
29	heat transfer in forced convection, effects of tube deformation on, 23:13942	英			

30	heat transfer method from coolant to working medium in reactor, 23:28538(P)				
31	operation of stainless steel, containing liquid potassium--sodium alloy, 23:13824(R) (IN-1201)	米			
32	performance of intermediate sodium, 23:24344(R) (LMEC-69-1)	米	IHX,SG	Na	LMFBR
33	performance of intermediate sodium, design of facilities for testing, 23:9610(R) (LMEC-68-13)	米		Na	LMFBR
34	performance of shell- and tube-type, structural and thermal, 23:40920 (UNC-5236)	米		Na	
35	power reactor add-on, description of multiple output, 23:23908(P)				
36	power reactor, list of suppliers for Europe, 23:5984	コーラトム			
37	radiation effects on dogs following implanation of plutonium-238 powered blood-cooled, γ, neutron, and thermal, 23:35892	米			
38	surfaces of reactor, method for minimization of carbon deposition on, 23:36179(P)				
39	thermodynamic parameters for carbon dioxide cooled power reactor, tabulation of, 23:28545	スイス	reheater	CO_2	HTGR
40	tube bundles for, design of guide vanes for, 23:13916(P)				
41	tube sheets for, analysis of thermal stresses in U-tube-type, 23:38141 (KAPL-P-3837)	米			
42	tube spacers for, design of bored plate, 23:11699(P)				
43	tube vibration in reactor, review of data on flow-induced, 23:24348 (ORNL-4399)	米			
44	tube vibration in, engineering method for evaluation of, 23:26405 (GA-8292)	米	SG		
45	tube welding in, electric arc method for, 23:13915(P)				
46	tubes for, design of ribbed thermoplastic, 23:47707(P)				
47	tubes in, description of two-phase fluid flow in vertical, 23:21878 (CONF-680118-1)	英	SG evaporator		
48	tubesheets for power reactor, fatigue testing of, 23:31189	カ			
49	tubing materials for nuclear power plant, selection and performance of, 23:22457	スウェーデン		水	LWR
50	HEAT EXCHANGERS (GAS-GAS) description of gas-cooled reactor multiple parallel tube-bend-connected finned-tube, 23:9144(P)				GCR
51	description of gas-cooled reactor multiple parallel tube-bend-connected finned-tube, 23:9140(P)				GCR
52	HEAT EXCHANGERS (GAS-LIQUID) configuration of gas-cooled reactor vertical-axis bell-shaped forced-flow once-through tube-and-shell, description of, 23:2028(P)				GCR
53	gas-cooled reactor grate supported multiple element, description of closed loop, 23:23907(P)				GCR
54	removal from gas-cooled reactor concrete pressure-vessel, description of, 23:15607(P)				GCR
55	safety problems of high temperature graphite moderated gas cooled power reactor, 23:37490	仏			GCR
56	HEAT EXCHANGERS (LIQUID-LIQUID) design for BN-350 Reactor, 23:52430(T) (JPRS-48330, pp 252-68)	ソ	IHX SG	Na-Na Na-水	BN-350(FBR)
57	design for improved heat transfer from liquid sodium, 23:17647(P)			液体金属	

58	design for separate fluid streams, 23:45466(P)			液体金属	
59	design parameters for 1000 MW(e) single-fluid MSBR vertical single-pass counterflow, 23:21452 (ORNL-4344, pp 53-108)	米		液体-液体	MSBR
60	heat transfer in FFTF primary loop tube and shell, multi-node model analysis of, 23:23954 (BNWL-CC-664)	米	IHX	Na	FFTF
61	heat transfer in liquid metal-cooled reactor countercurrent liquid metal double-pipe, (E/T), 23:19486 (ANL-7479)	米		液体金属	
62	operation dynamics for MSBR salt--supercritical water, use of analog computer for, 23:13486 (ORNL-TM-2405)	米		水	MSBR
63	operation performance of EBR-2, Fermi Fast Breeder Reactor, SRE, and Hallam Power Reactor, review of, 23:11193 (CONF-680419-, pp 326-53)	米	IHX		EBR-2,Fermi SRE,Hallam
64	safety problem analysis for liquid sodium-cooled fast power reactor single-wall sodium and water, 23:17183(T) (CONF-670916-4)	ソ		Na-水	LMFBR
65	tube failures in HFIR reactor intermediate, effects of hydrodynamic vibration on, 23:19529 (ORNL-TM-2467)	米	IHX		HFIR
66	tube failures in HFIR, analysis of vibration induced, 23:23974(R) (ORNL-TM-2527)	米	IHX		HFIR
67	tube failures in sodium heated, pressure pulses from sodium--water reactions following, 23:28536 (EUR-4101, pp 125-42)	西独	SG	Na-水	
68	tube leaks in sodium heated, hazards from sodium--water reaction following, 23:28507 (EUR-4101, pp 75-88)	英	SG	Na-水	

1970 Vol. 24

70000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators			
1	assembly and dismantling of elements of, design of equipment for, 24:9553(P)			
2	book: Heat Exchangers for Nuclear Power Installations, 24:18753	ソ		
3	corrosion by seawater, testing of, 24:30978	仏		
4	decontamination of surfaces of stainless steel, chemical methods for, 24:19063(P)			
5	description of copper foil, for dilution refrigerators, 24:35348	仏		
6	design and testing of double-wall, for fluidized bed volatility process plant, 24:2344	SG 米		
7	design for nuclear reactor, 24:46407(P)			
8	design of liquid metal, with buffer vessel for safety, 24:7569(P)			
9	design of liquid sodium heated, with flow paths for prevention of tube dry out, 24:36058(P)		Na	
10	design of power reactor, problems and requirements for standardization of, 24:18607	伊		
11	design of single fluid phase regenerative, for nuclear power plants, 24:19130	西独		
12	design of tubular, contained in pressure vessel, 24:46408(P)			
13	design parameters for carbon dioxide cooled power reactor regenerative, 24:11181(P)		CO_2	GCR
14	design parameters of bayonet tube, 24:47810 (ORNL-4548, pp 41-92)	米		MSBR
15	design parameters of removable integrated, for gas cocled or boiling reactors, 24:52911(P)			GCR BWR
2	design requirements for marine and power reactor, 24:18753	ソ		マリン
17	development for temperature control in heat conveying fluids, 24:25191(P)			
18	development in Netherlands for liquid metal cooled fast reactors, 24:52896	オランダIHX,SG	液体金属	LMFBR
19	development of liquid sodium, CNEN program for, 24:16226	伊	Na	
20	efficiency of regenerator, influence of thermal conduction on, 24:36101(T) (ANL-Trans-837)	西独		
21	energy accumulation in packing of regenerative, 24:41077(T) (ANL-Trans-836)	ポーランド		
22	fabrication of Dodewaard Power Reactor steam condenser, joining of tubes to tube sheet in, 24:11108	オランダ	水	Dodewaard (BWR)
23	fabrication of porous copper, for He-3--He-4 dilution refrigerator, 24:15200 (LA-4259)			
24	heat transfer in counter flow and parallel flow, frequency method for dynamic analysis of, 24:36495	ユーゴ		
25	modeling for FFTF simulation, methods for, 24:31578 (BNWL-1367)	米 IHX	Na	FFTF
26	noise spectra from coolant flow in Dragon Reactor, (E), 24:4083 (AEEW-R-641)	英	He	Dragon (HTGR)

27	performance in graphite moderated carbon dioxide cooled power reactors, 24:28931	仏		CO_2	Chinon 1.2.3 (GCR)
28	performance of carbon dioxide cooled power reactor, 24:28867	仏		CO_2	EL4 (GCR)
29	performance of MSRE primary, summary of, 24:47809 (ORNL-4548, pp 1-40)	米			MSRE
30	performance testing of prototype LMFBR, 24:20936(R) (LMEC-70-1)	米		Na	LMFBR
31	pressure casing for Dragon Reactor, metallurgical properties of, 24:39623 (DP-Report-80)	英		He	Dragon (HTGR)
32	properties of flat-tube continuous-fin, 24:36100(T) (ANL-Trans-835)	西独			
33	simulation of fluid-cooled, development of equipment for, 24:36499	ベルギー			
34	simulation of forced circulation, development of analog system for, 24:36500	伊			
35	simulation of reactor forced circulation, method for hybrid computer, 24:38534	米			
36	simulation of thick annular, analog computer method for, 24:36498	ベルギー			
37	simulators for, models for hybrid, 24:43808 (BNWL-SA-2818)	米	IHX	Na	FFTF (高速炉)
38	sodium boiling in, effects of pressure and temperature on incipient, 24:2384	米			
39	steam--water flow in fog-cooled reactor power plant, dynamic model for, 24:2371 (CISE-R-121)	伊		蒸気	
40	testing for use in LMFBR-type reactors, 24:48796(R) (LMEC-70-13)	米	SG	Na	LMFBR
41	testing of helically wound, for THTR steam generator, 24:36455 (JUL-649-RB)	西独	SG	air(test) He	THTR
42	testing of vertical counterflow, for LMFBR design data, 24:14126(R) (LMEC-69-30)	米	SG	Na	LMFBR
43	tube plugging in, design of equipment for defective, 24:48213(P)				
44	tube plugging method for, description of remote explosive, 24:46038	米			
45	tubes for steam power plant, fabrication and properties of stainless steel, 24:51167				
46	tubes for, analysis of cross-flow-induced vibrations in, 24:52875 (ANL-7685, pp 205-24)	米			
47	tubes for, design of remote handling equipment for maintenance and welding of ends of, 24:36462(P)				
48	tubes for, detection of flow induced vibrations in, 24:52874 (ANL-7685, pp 177-204)	米	IHX	Na	
49	HEAT EXCHANGERS (GAS-GAS) friction and heat transfer for single elements in Dragon Reactor helium purification, measurement of, 24:39132 (DP-Report-126(Pt.II)	英		He	Dragon(HTGR)
50	heat transfer in Dragon Reactor helium purification, theoretical studies of, 24:39131 (DP-Report-126(Pt.I)	英		He	Dragon(HTGR)
51	HEAT EXCHANGERS (GAS-LIQUID) control system for liquid sodium--air, description of air flow, 24:31098 (CEA-N-1263)	仏		Na-air	Rapsodie(LMFBR)
52	design of, 24:52953 (DP-Report-222)	英		He	Dragon(HTGR)
52	fabrication of, 24:52953 (DP-Report-222)	英		He	〃

52	heat transfer characteristics of, (E), 24:52953 (DP-Report-222)	英	He	Dragon(HTGR)
53	heat transfer in Dragon reactor, safety margin for, 24:39130 (DP-Report-113)	英	He	Dragon(HTGR)
54	heat transfer in -team generating, for gas cooled reactors, (E), 24:16227	西独 SG	GCR	GCR
54	hydrodynamic resistance steam generating in, for gas cooled reactors, (E), 24:16227	〃	〃	GCR
55	operation of Dragon reactor, corrosion and fretting problems in, 24:18674(R) (NP-18148)	英		Dragon(HTGR)
56	simulation of liquid sodium-to-air dump, dynamic digital computer, 24:26802 (BNWL-1319)	米 RHX	Na-air	FFTF (高速炉)
57	HEAT EXCHANGERS (LIQUID-LIQUID) design parameters of intermediate, for sodium cooled fast reactors, 24:50174 (WARD-4112-2(Vol.1))	米 IHX	Na-Na	FFTF(LMFBR)
58	development of liquid sodium cooled, 24:43689 (CONF-700307-5)	仏	Na-Na	Rapsodie Phenix (LMFBR)
59	flow control in liquid metal cooled reactors, description of baffle system for, 24:22402(P)		液体金属	
60	maintenance of liquid sodium cooled reactor, review of, 24:5524	西独	Na	
60	operation of liquid sodium cooled reactors, review of, 24:5524	西独	Na	
61	performance of intermediate, for sodium cooled fast reactors, (T), 24:50175 (WARD-4112-2(Vol.2)	米 IHX	Na	FFTF(LMFBR)
62	repair device for sodium, design of, 24:39113 (NP-18292)		Na	
63	testing program for liquid sodium intermediate, 24:52903	日 IHX,SG	Na- , Na-Na	
64	testing program for liquid sodium intermediate, 24:52902	米 IHX,SG	Na- Na	

1971 Vol. 25

71000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators				
1	analysis of, for dilution refrigerators, 25:45441	米			
2	carbon deposits on surfaces of reactor, method for prevention or reduction of, 25:24564(P)				
3	description of intermediate and secondary, for sodium-cooled fast breeder reactors, 25:18574	日	IHX	Na	FBR
4	design and development for gas turbine fast reactor plants, review on, 25:23724 (EIR-187)	スイス			高速炉
5	design and fabrication of improved helical-type, 25:57630(P)				
6	design and performance of plate, for secondary cooling system of nuclear power plants, 25:29269	スウェーデン			
7	design for use in preparation of radioactive tracer samples, 25:21252(P)				
8	design in Switzerland of water reheater, for power reactors, 25:8072	スイス	reheater water separator		
9	design of filter plenum, for radioactive processing buildings, 25:54950 (CONF-710401-, pp 162-80)	米			
10	design of molten salt breeder reactor, computer programs for, 25:25974 (ORNL-TM-2815)	米	IHX reheater SG		MSBR
11	design of reactor auxiliary feedwater, connected to de-aerators, 25:51118(P)				
12	design parameter for gas cooled reactors, with means to allow for expansion of tubes, 25:12811(P)				GCR
13	design parameters for direct cycle gas turbines, 25:20775	スイス			
14	design parameters of module, for power reactors, 25:10204(P)				
15	design parameters of steam reheater, with U-shaped tube banks in concrete pressure vessel walls, 25:12810(P)				
16	design requirements for direct cycle gas turbines, 25:20776	英			
17	design with low cost materials, description of, 25:46160(P)				
18	heat resistance in He-3--He-4 dilution refrigerator, proposal to decrease, 25:45436	オランダ			
19	heat transfer and temperature distribution in plane cross flow, with heat generation in one fluid, method for calculation of, 25:29262	伊			
20	leaks in welded nuclear power plant, comparison of ammonia and helium test methods for detecting, 25:6502	仏			
21	numerical analysis of continuous and discrete, for dilution refrigerators, 25:45433	米			
8	operation in Switzerland of water reheater, for power reactors, 25:8072	スイス	reheater water separator		
22	optimization code for, 25:44265 (K-1799)	米			
23	performance of large component test facility sodium, evaluation of, 25:42804(R) (LMEC-71-1)	米	IHX	Na	
24	performance of tubular, effects of twisted tapes on thermal, 25:57643	伊			PWR LMFBR

25	sodium boiling on surfaces of, effects of heating surface material and roughness on incipient, 25:54868(T) (EURFNR-897)	西独		Na	
26	stability of materials in MSRE, evaluation of, 25:58168 (ORNL-4676, pp 139-233)	米			MSRE
27	standard for design and performance of air-cooled, for nuclear steam-supplied system, 25:54833 (RDT-E-4-18-T)	米		air	
28	standard for fabrication of liquid metal-to-liquid metal intermediate, 25:39629 (RDT-E-4-6-T)	米	IHX	液体金属	
29	standard for seamless annealed nickel--chromium--iron alloy tubes for, 25:42782 (RDT-M-3-4-T)	米			
30	supports for, description of rotating, 25:20840(P)				
31	temperature control in internal heat-generating, method for optimum, (E/T), 25:26421	米			
32	temperature distribution in molten salt reactor, calculation of fission product decay heat, 25:41413	米			MSRE
33	temperature stabilization in, under variable heat flux, use of heat pipes for, (E/T), 25:34775 (EUR-4634)	伊			
34	thermodynamics of tube type, FORTRAN 4 computer programs for calculation of, 25:20835 (RT/ING-(70)8)	伊			
35	tube dryout with boiling water at 350 psi, forced circulation, (E), 25:48246 (AEEW-M-940)	英			
36	tube joints for sodium heated, x-ray and ultrasonic methods for inspection of, 25:44148 (CONF-700224-, Paper 4)	オランダSG		Na	
37	tubes for, review on nonferrous metal, 25:45060	伊			
38	tubes for, vibration of U-bend segments of, 25:48689 (AECL-3735)	カ			
39	tubes in, explosive plug sealing of leaking, 25:13647	米			
40	welds in reactor, methods for in-service inspection and repair of, 25:51184	米			
41	HEAT EXCHANGERS (GAS-GAS) development of dry air, for direct cycle gas turbine helium coolec power reactors, 25:2077	西独		He	
42	gas outlet temperatures of reactor counter-flow, effects of step change of inlet temperatures on, (T), 25:20779	西独			
43	tube vibration in, prevention of, 25:23230	日			GCR
44	HEAT EXCHANGERS (GAS-LIQUID) design and performance of forced-flow bare tube, computer program for analysis of, 25:48699 (WERL-TAGOV-1)	米			
45	design characteristics of secondary, for gas cooled reactors, 25:60074(P)				GCR
46	heat transfer performance of high temperature reactor, in high pressure helium, (E), 25:20780	西独		He	HTR
47	HEAT EXCHANGERS (LIQUID-LIQUID) baffles for heavy water cooled reactor primary, analysis and repair of damage to, 25:37055 (BNL-15815)	米		重水	HWR
48	construction of liquid sodium cooled fast reactor, 25:54071 (A/CONF.49/P-482)	英	IHX	Na	LMFBR
49	design evaluation of sodium intermediate reactor, using austenitic or ferritic steel, 25:57614 (KFK-1387)	西独	IHX	Na	
48	design for liquid sodium cooled fast reactors, 25:54071 (A/CONF.49/P-482)	英	IHX	Na	LMFBR
50	design for use in helium isotope separation, 25:13165(R) (MLM-1753)	米			
51	design of liquid metal heated tubular, with means for prevention of localized overheating, 25:28749(P)			液体金属	

52	design problems with liquid sodium cooled fast reactor intermediate, 25:60077	仏	IHX	Na	Phenix(LMFBR)
53	development in Japan of intermediate liquid metal cooled, for fast breeder reactor, 25:18133	日	IHX,SG	Na	LMFBR
54	heat transfer in sodium--water counterflow, model for analysis of, 25:6463 (CEA-CONF-1592)	仏		Na-水	
55	leak detection system for sodium-water, development of hydrogen monitoring, 25:50903	英		Na-水	高速炉
56	leak in sodium/water, sound speed and choking onset in nonequilibrium two-phase flow from large, (T), 25:48716	英		Na-水	
57	standard for fabrication and acceptance of shell and tube, 25:54832 (RDT-E-4-17-T)	米		水－水	

1972 Vol. 26

72000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators				
1	arrangement of steam reheater, outside pressure vessel of gas cooled power reactor, 26:3939(P)				GCR
2	corrosion by two-phase flowing water, 26:33893 (INIS-mf-213)	仏			
3	corrosion of Dragon reactor, review of experience on, 26:449	英		He	Dragon(HTGR)
4	damage to, effects of sodium--water reactions on, 26:55158 (CONF-710548-, pp 187-205)	英		Na-水	高速炉
5	description of improved high-pressure, 26:9100(P)				
6	design and performance of sodium-to-air, for cooling Fast Flux Test Facility, review on, 26:9072 (AI-AEC-Memo-12869)	米		Na-水	FFTF (高速炉)
7	design characteristics of annular, for radial coolant flow in gas cooled reactors, 26:37826(P)				GCR
8	design characteristics of thermal convection liquid--liquid, for two coolant region liquid sodium cooled breeder reactor, 26:24822(P)			Na	LMFBR
9	design criteria for, for use with vapor generating equipment, 26:45436 (CEA-CONF-2010)	仏	SG		
10	design for power plant using gas-cooled breeder reactor and gas turbine, parametric study of, 26:45439 (KFK-1566)	西独			GCFR
11	design for sodium-cooled reactors, 26:238	英		Na	
12	design for use in liquid metal-heated steam generators, 26:22715(P)	英	SG	液体金属	
13	design of concentric-pipe, for liquid sodium systems, p.2-33, 26:31450(R) (LMEC-72-1)	米	SG	Na	FFTF(高速炉) LMFBR
14	design of condenser cooling system, for optimum performance of steam power plant, 26:25449	米			
15	design of high-efficiency, using tubes with turbulence-promoting inner surfaces and external surface grooves, 26:36015	ソ			
16	design of improved U-tube, 26:2033(P)				
17	design of improved, with means for cooling of casing, 26:231(P)				
18	design of liquid metal, with thermal barrier, 26:25443(P)				
19	design of minimum volume, for safe heat transfer in expensive or dangerous fluids, 26:9098(P)				
20	design of modular-unit, for gas- or liquid-cooled reactor, 26:11918(P)				GCR
21	design of parallel plate, for use in annulus surrounding reactor fuel element, 26:6749(P)				
22	design of plate type, with anticlastic surfaces, 26:24994(P)				
23	design of sodium--sodium intermediate, for fast breeder reactors, development of SINEX computer program for calculating optimal parameters for, 26:58731	西独	IHX	Na	FBR
24	design of tube-in-shell-type, for reactors, 26:58732(P)				
25	design of water-cooled reactor, for reheating part of steam, 26:24785(P)				LWR

26	design parameters for Brayton cycle operation space vehicle power reactor, calculation of optimum, 26:6346 (N-71-14037)	米			ブレイトンサイクル
27	design parameters of liquid-liquid, for molten salt reactors, optimum, 26:6258 (ORNL-TM-2952)	米			MSRE
28	development for MHD generator research, 26:56023 (FRNC-TH-231)	仏			MHD
29	dismantling of sodium cooled reactor, removal of sodium and decontamination during, 26:55147 (AI-AEC-Memo-12736)	米		Na	Hallam
30	economics of sodium cooled reactor intermediate, effects of costs of austenitic versus ferritic steels on, (T), 26:1566(T) (EURFNR-924)	西独	IHX	Na	
31	economics of sodium cooled, effects of ferritic steels use on, 26:42514	西独	IHX	Na	
32	failure of tubing in molten salt breeder reactor, analysis of coolant and fuel mixing from, 26:24754 (ORNL-4728, pp 1-44)	米	SG		MSBR
33	flow model for counter current sodium, hybrid computer program for, 26:60598	オランダ		Na	
34	gas flow in closed circuit, with variable operating conditions, numerical method for simulating dynamic and thermal characteristics of, 26:25467	仏			
35	gas removal from steam reheater, method for noncondensable, 26:22213(P)				
36	gas-liquid heat transfer and pressure drop in, with cross flow over bare tubes, 26:40771	仏			
37	heat transfer rates in double-pipe sodium-to-sodium counter-flow, effects of inner pipe placement eccentricity on, (E), 26:25465	米		Na	
38	leaks of water into liquid sodium in, effects on temperature and pressure of, 26:55152 (CONF-710548-, pp 39-56)	ソ			
39	materials for, fabrication and properties of brazed titonium alloys cs, 26:15783	米			
40	performance of copper-foil, in dilution refrigerators, 26:13353	仏			
41	performance of gas cooled reactor, with coiled tubes, (E), 26:42512	西独	SG		GCR
42	performance of, use of analytic nodal average temperatures in simulation of transient, (T), 26:45005	米			
43	pressure containment tests for Brayton cycle operation space vehicle power reactor, (E), 26:6347 (N-71-16560)	米			ブレイトンサイクル
44	repair of loose tubes in fast reactor intermediate, 26:19707 (ANL-7834)	米	IHX	Na	EBR-II(高速炉)
45	selection for use as gas turbine regenerators in direct-cycle nuclear power plants, thermodynamic studies for, 26:25466	仏			
46	sodium aerosol deposition on surfaces of, equipment and method for measuring, 26:48483(R) (COO-3019-1)	米		Na	
47	sodium deposition in cover gas, (E), 26:52355(R) (COO-3019-1)	米			
48	standards for intermediate, for sodium cooled reactors, 26:35411 (RDT-E-4-6-T(1-72)	米	IHX	Na	LMFBR
49	standards for reactor, for cooling circulating inert gas or air, 26:49997 (RDT-E-4-20-T(5-72)	米			
50	survey of LMFBR intermediate, 26:58317	英	IHX,SG	水, Na	LMFBR

51	temperature distribution in counter-flow liquid potassium--sodium alloy, computer analysis of, 26:65026 (SFB-MHD-3)	西独		Na	
52	testing of large-scale, development of equipment for, 26:56015	仏	SG		
53	tube banks in, effects of gas vortex shedding on vibration of, (E), 26:27391	日			
54	tube vibrations in liquid sodium cooled, 26:30207	仏	IHX	Na	Phenix(FBR)
55	tubes for air cross-flow-type, development of improved heat transfer surfaces for, 26:25469	英			AGR,THR
56	use of segmented, in liquid metal system, schematics for, 26:25438 (CNLM-3294)	米		液体金属	
57	vibrations in carbon dioxide heated, effects of configuration and operating conditions on, (E/T), 26:22234	仏		CO_2	EDF-3(GCR)
58	vibrations in gas-liquid, characteristics of acoustic, (E/T), 26:22233	仏		CO_2	Chinon-3(GCR)
59	vibrations in water cooled reactor, analysis of flow induced, 26:44660	米		水	LWR

1973 Vol. 27

73000	HEAT EXCHANGERS see also Boilers see also Heat Transfer see also Radiators				
1	cold traps for liquid metal, design of, 27:6924(P)	英			
2	description of in vessel, for sodium cooled reactor, 27:4425(P)	仏	IHX	Na	LMFBR
3	design and performance of, for radioisotope-fueled Brayton power system for space vehicles, 27:9791	米			ブレイトンサイクル
4	design for high-temperature gas-cooled reactors, anti-fretting, 27:14042	英		He	HTGR
5	design for sodium-cooled reactors, 27:11394(P)	西独		Na	LMFBR
6	design of heat pipe heated, for use in Brayton cycle space power system, 27:4723(R) (LA-4625)	米			ブレイトンサイクル
7	design of liquid metal, for cooling radiation targets, 27:2485(P)	仏			
8	design of liquid sodium--water, 27:11392(P)			Na-水	
9	design of regenerative helium, for use with gas cooled breeder reactor and gas turbine generator, 27:4454 (EURFNR-1010)	西独	RHX	He	GCFR
10	design of silicon tube, for transfer of heat from reactors to iron ore reduction processes, 27:15210	西独			
11	design of single shell counter flow, with non-equal tube passes, (T), 27:16510 (AREAEE-144)	エジプト			
12	design of sodium, hydraulic problems in, 27:6918 (CEA-CONF-2046)	仏		Na	
13	design of sodium/sodium intermediate, using SINE computer code, 27:23642 (EURFNR-1052)	西独	IHX	Na	
14	design of sodium-cooled in-vessel, for fast reactors, 27:11396(P)			Na	高速炉
15	design of, computer program for optimum, 27:22410	スイス			
16	design of, relations of mechanical safety with power ratings in, 27:11919 (CONF-720849-1)	米			
17	design optimization of sodium/sodium, computer program for, 27:14651 (KFK-1628)	西独	IHX	Na	
18	design parameters for sodium/sodium intermediate, computer program for optimization of mechanical and thermal, 27:22408 (KFK-1686)	西独	IHX	Na	
19	development of indicrect, for transfer of nuclear heat to reduction processes, 27:14660	西独			
20	development of integrated steam generator--, for sodium cooled fast breeder reactors, 27:23904 (BMBW-FBK-72-28)	西独	IHX	Na	LMFBR
21	development of liquid metal-cooled, for high-temperature nuclear technology, 27:17195	西独		液体金属	HTR
22	effects of sintered copper, on superfluid helium viscosity measurements, 27:23456	英			
23	fabrication of liquid sodium, using electron-beam welding machine, 27:20304	仏		Na	
24	flow instabilities in gas-liquid, effects of configuration on, 27:29468 (DP-Report-651)	英		He	Dragon(HTGR)
25	flow patterns in inlet and outlet of sodium, (E/T), 27:6918 (CEA-CONF-2046)	仏		Na	

26	heat transfer surfaces in compact, evaluation of efficiency of, 27:2480 (K-L-6295)	米			
27	installation in walls of sodium cooled reactor vessels, device for sealed, 27:4778(P)			Na	
28	leak location in tubes of sodium--water, method for, 27:11393(P)			Na-水	
29	materials development program for sodium-cooled fast reactors, October 1971 to August 1972, 27:26755(R) (CEAP-13919-1)	米		Na	LMFBR
30	performance characteristics of geometrically different, method for obtaining and reducing data on, 27:11920 (K-1832)	米			
31	performance in vapor generators, effects of surface deposits on, 27:5017		SG		
32	pressure drop in shell and tube-type, diffusion-controlled electrochemical mass transfer modeling method for predicting, 27:14340 (AEEW-R-809)	英			
25	pressure losses at inlet and outlet of sodium, (E/T), 27:6918 (CEA-CONF-2046)	仏		Na	
33	repair of tubular array, machine for, 27:9137(P)				
34	response of tubular, to inlet fluid temperature variations, calculation of dynamic, 27:17194	伊			
35	stress analysis of, development of TRANS computer program for dynamic, 27:2152	オーストリア	SG		
36	testing of helical, for gas-cooled reactors, 27:14041				GCR
37	tube erosion in sodium heated, model for effects of small water leaks on, (E/T), 27:1981	英		Na-水	
38	tube support wear in primary, design studies for reducing viorational effects on, 27:22412	スイス			HTR
39	tube testing for, eddy current method for, 27:14663	西独			
40	tubes for, properties and fabrication requirements for finned, 27:2204	西独			
41	tubes of intermediate, single-frequency eddy-current testing of sodium contaminated, 27:24172 (HEDL-TME-72-152)	米	IHX	Na	
42	vibrations in carbon dioxide heated, effects of configuration and operating conditions on, (E/T), 27:23874(T) (LIB/Trans-482)	仏		CO_2	EDF-3(GCR)

1973 Vol. 28

73000	HEAT EXCHANGERS See also IN-VESSEL HEAT EXCHANGERS RADIATORS				
43	Graphite as a material for a heat exchange for a hot gas (Patent; covering of borocarbide, silicide, or oxide), 28:13166				
44	Steam generator and heat exchanger materials (LMFBR), 28:8624 (ORNL-TM-4105)	米			LMFBR
45	HEAT EXCHANGERS/CONTAMINATION Calculation of the circuits activity in a high temperature reactor, 28:7285 (CEA-CONF-2214)	仏			HTR
46	HEAT EXCHANGERS/CORROSION Corrosion problems related to two-phase flow, 28:8100	仏			Marcoule Chinon-2
47	HEAT EXCHANGERS/COST Cost analysis of components for nuclear power plants with high-temperature reactor and gas-turbine, 28:7296 (JUL-920-RG)	西独		He	HTR
48	HEAT EXCHANGERS/DEFORMATION Plastic strain analysis of heat exchanger tubesheets, 28:14729 (EUR-4820(Vol.3)(Pt.F)	米			
49	HEAT EXCHANGERS/DEPOSITS Particle deposition and reentrainment on FFTF H and V components following a sodium spill, 28:31854 (CONF-720823-P1)	米		Na	FFTF (高速炉)
50	HEAT EXCHANGERS/DESIGN Heat exchanger element, 28:5465 (Patent)				
51	Heat exchanger module, 28:20302 (P)				
52	High temperature regenerative heat exchangers for open cycle MHD power plant. II, 28:27160	日	RHX		MHD
53	Improvements in or relating to nuclear reactor power plants, 28:17578 (P)	英			
54	Main components in high power fast breeder reactors, 28:31829	仏	IHX,SG	Na	FBR
55	Safety criteria for a heat exchanger in the primary circuit of a sodium cooled power reactor, 28:10017	伊		Na	
56	SINEX: a code for thermodynamic and strength design calculation of sodium-heated intermediate heat exchangers, 28:20371 (EURFNR-1075)	西独	IHX,	Na	
57	Solar collector thermal power system: its potential and development status, 28:20937	米			
58	HEAT EXCHANGERS/FABRICATION Improvements to tubular corrosion-resistant heat exchangers, 28:18165 (Patent)	仏			
59	Steam generator and heat exchanger materials, 28:11218 (ORNL-TM-4055)	米	SG		
60	HEAT EXCHANGERS/FINS Turbulent conductivity in a flow between two plane parallel walls in a regime established with a transverse velocity gradient, 28:17733 (CEA-R-4442)	仏			
61	HEAT EXCHANGERS/GAS FLOW Experimental investigation of air recirculation characteristics of air cooled condenser arrays on a process building. Report EERL-TR--105, 28:5454 (GAT-Z-4128(Vol.2))	米			
62	HEAT EXCHANGERS/HEAT TRANSFER Structure of sodium corrosion deposits and their effect on heat transfer coefficients, 28:15971	米	IHX,SG	Na	

63	HEAT EXCHANGERS/NOZZLES High-temperature stress analysis program (LMFBR), 28:26513 (GEAP-13944)	米	IHX	Na	LMFBR
64	HEAT EXCHANGERS/PERFORMANCE Heat transfer and pumping power performance in tube banks: finned and bare, 28:15662 (CONF-730803-3)	米			
65	High temperature regenerative heat exchangers for open cycle MHD power plant. I, 28:18102	日	RHX		MHD
66	Solar collector thermal power system: its potential and development status, 28:20937	米			
67	TUMHEX: a computer code for the calculation of the frequency response behavior of shell-and-tube type counterflow heat exchangers with respect to primary and secondary inlet temperature and mass flow perturbations, 28:15435 (MRR-P-2)	西独			
68	HEAT EXCHANGERS/PERFORMANCE TESTING Heat transfer tests on intermediate heat exchanger model for LMFBR (LMFBR), 28:17445	日	IHX	Na	LMFBR
63	HEAT EXCHANGERS/PIPE JOINTS High-temperature stress analysis program (LMFBR), 28:26513 (GEAP-13944)	米	IHX	Na	LMFBR
69	HEAT EXCHANGERS/SPACERS Spacing devices for a tubular array: heat exchange utilizing these devices (LMFBR), 28:17442 (Patent)			Na-水	LMFBR
70	HEAT EXCHANGERS/SPECIFICATIONS Air-cooled heat exchanger performance specification, 28:32139 (AERE-R-7450)	英		air	
71	HEAT EXCHANGERS/STANDARDS Heat exchanger, class 1, water-to-water, straight or U-tube, 28:17575 (RDT-E-4-2-T(5-73))	米		水－水	
72	Report on current standards for air-cooled heat exchangers, 28:12804 (TID-26324)	米		air	
48	HEAT EXCHANGERS/STRAINS Plastic strain analysis of heat exchanger tubesheets, 28:14729 (EUR-4820(Vol.3)(Pt.F))	米			
73	HEAT EXCHANGERS/STRESS ANALYSIS Errors in stress due to errors in strain gage readings, 28:12800 (CONF-721089-3)	米		Na-Na Na-水 蒸気－蒸気	
74	HEAT EXCHANGERS/TEMPERATURE CONTROL Improvements in or relating to nuclear reactor power plant (Load control), 28:1779 (P)				
75	HEAT EXCHANGERS/THERMAL CONDUCTION Treatment of three-dimensional heat conduction and thermoelastic problems by the finite element method, 28:10799 (EUR-4820(Vol.6)(Pt.M))	西独			
76	HEAT EXCHANGERS/TUBES Holding spacer for nuclear-fuel rods, etc., 28:10188 (JAERI-M-4758)	日			
77	HEAT EXCHANGERS/WELDED JOINTS High integrity tube-to-tubeplate bore welds, 28:23499	英			

1974 Vol. 29

74000	HEAT EXCHANGERS See also HEAT PUMPS IN-VESSEL HEAT EXCHANGERS RADIATORS				
1	Formulas for single-phase convection heat exchanges: effect of geometry, physical properties, and method for improving the heat exchange, 29:7415 (CEA-N-1660)	仏			
2	Heat exchanger (Patent), 29:17582				
3	Two regimes of a spray column in countercurrent flow, 29:28932 (CEA-CONF-2499)	仏			
4	HEAT EXCHANGERS/CERAMICS Use of nonmetallic materials with high-temperature gas-cooled reactors with special reference to heat exchangers for power and process plants, 29:11790 (CONF-730668-1)	英		He	HTGR
5	HEAT EXCHANGERS/CONFIGURATION Heat exchange installation for a nuclear reactor (Patent), 29:12154				
6	Heat exchanger (Patent), 29:14651				
7	HTGR gas turbine power plant configuration studies, 29:31193 (GULF-GA-A-12651)	米		He	HTGR
8	Improvements to plate heat exchangers and thermal energy producing installations having such exchangers (Patent), 29:11786				
9	HEAT EXCHANGERS/CONVECTION Some current problems in convective heat transfer, 29:10085	ソ			
10	HEAT EXCHANGERS/CORROSION LMFBR heat exchanger materials development program. Fourth quarterly progress report, March 1973--June 1973, 29:9361 (GEAP-13919-4)	米		Na	LMFBR
11	HEAT EXCHANGERS/DESIGN Computer-aided design and optimization of heat exchangers, 29:20275	西独	IHX	Na	
12	Conceptual design of a food complex using waste warm water for heating, 29:2651	米			
13	Design, fabrication, and test of a 1500F NaK--NaK heat exchanger, HE-14S, 29:26797 (PWAC-368)	米		NaK	
14	High temperature helium heat exchanger loop, 29:14690	日		He	
15	Improvements in or relating to heat exchangers (Patent), 29:18269				
16	Improvements to tubular heat exchangers (Patent), 29:29231				
17	Problems of heat transfer and the design of high-temperature heat exchangers for electrical power plants having MHD-generators, 29:244	ソ			MHD
18	SIMPLEXONE controller for optimization studies, 29:9516 (AECL-4219)	カ			
19	HEAT EXCHANGERS/FABRICATION Fuels and Materials Development Program quarterly progress report for period ending March 31, 1973, 29:966 (ORNL-TM-4250)	米		Na	FBR
20	Intermediate heat-exchanger (na--Na) (LMFBR), 29:23518	インド	IHX	Na-Na	LMFBR
21	HEAT EXCHANGERS/FLOW RATE LMFBR problems due to large temperature differences, 29:6945	米		Na	LMFBR

22	HEAT EXCHANGERS/FLUID FLOW Determination of eddy diffusivities in tube bundles with cross flow, 29:7676	スイス			
23	HEAT EXCHANGERS/FRETTING CORROSION Impact fretting of heat exchanger tubes, 29:8253 (AECL-4653)	カ			
24	HEAT EXCHANGERS/GEOTHERMAL ENERGY Evaporative heat transfer in vertical tubes at geothermal brine conditions: a preliminary investigation, 29:20413 (ORNL-TM-4135)	米			
25	HEAT EXCHANGERS/HEAT TRANSFER Mean temperature difference of multichannel heat exchangers, 29:12148	ソ			
26	HEAT EXCHANGERS/HYDRODYNAMICS Hydrodynamic oscillations in parallel-fed heating channels, 29:10066 (CEA-CONF-2364)	仏			
27	HEAT EXCHANGERS/LEAK TESTING Method of locating a steam leak in the circulation heat exchanger immersed in liquid sodium and the corresponding location equipment, 29:7672	仏	Na		
28	HEAT EXCHANGERS/MASS TRANSFER Linear analytical one-dimensional model describing the frequency response behavior of shell-and-tube type counterflow heat exchangers with respect to primary perturbations, 29:29223	西独			
29	HEAT EXCHANGERS/MECHANICAL STRUCTURES Device for the leak-tight assembly of heat exchangers nuclear reactors (Patent), 29:17594				
30	HEAT EXCHANGERS/MECHANICAL VIBRATIONS Industrial experience of vibration problems in thermal equipment, 29:29234	仏			
31	Mechanical vibration analysis of an integrated intermedium heat exchanger and circulating pump for LMFBR stations, 29:11873 (CONF-730942-(P2))	伊	IHX		LMFBR
32	Pressure distributions on in-line tube arrays in cross flow, 29:29237 (CONF-730465-)	英			
33	Study of flow and acoustic phenomena in a tube-bank, 29:31176 (CONF-730465-)	仏			
34	Vibration of cylindrical structures induced by axial flow, 29:29252	仏			
35	HEAT EXCHANGERS/MEETING Conference on recent developments in compact high duty heat exchangers, London, England, October 12, 1972, 29:12160				
36	HEAT EXCHANGERS/NONDESTRUCTIVE TESTING Fuels and Materials Development Program quarterly progress report for period ending March 31, 1973, 29:544 (ORNL-TM-4250)	米	Na		FBR
37	HEAT EXCHANGERS/OPTIMIZATION SIMPLEXONE controller for optimization studies, 29:9516 (AECL-4219)	カ			
38	HEAT EXCHANGERS/PERFORMANCE Heat transfer problems with nucleate boiling of liquids, 29:10084	ソ			
39	Plate heat exchangers in nuclear power stations, 29:23642	スウェーデン			
40	HEAT EXCHANGERS/PERFORMANCE TESTING Design fabrication and tests of a 1500°F NaK--NaK heat exchanger, 29:26166 (PWAC-337)	米	NaK-NaK		
13	Design, fabrication, and test of a 1500F NaK--NaK heat exchanger, HE-14S, 29:26797 (PWAC-368)	米	NaK-NaK		

41	Heat-transfer potential of liquid-gas spray flows, 29:18293					
42	Heating air and gases to 1500°C and higher in high-temperature regenerative heat exchangers with stationary ball packing, 29:245	ソ	RHX	air		
14	High temperature helium heat exchanger loop, 29:14690	日		He		
43	LCRE non-nuclear system test. Interim report, 29:23371 (PWAC-402(Pt.1))	米				
44	Methods for evaluating the performances of compact heat transfer surfaces, 29:10080	米				
45	Performance of compact heat transfer surfaces: finned and bare tube banks, 29:2600 (K-1847)	米				
46	HEAT EXCHANGERS/PERMEABILITY Multiple measurement techniques with process-computer control and data gathering at two test equipments for determining hydrogen permeation through heat-exchanger materials, 29:26785 (JUL-1020-RG)	西独				
47	HEAT EXCHANGERS/REACTOR VESSELS Device for the leak-tight assembly of heat exchangers in nuclear reactors (Patent), 29:17594					
48	HEAT EXCHANGERS/SAFETY STANDARDS Safety problems of future process-heat reactors, 29:7102	西独				
49	HEAT EXCHANGERS/SODIUM Heat exchangers for sodium-cooled nuclear reactors (Patent), 29:17723			Na		
50	HEAT EXCHANGERS/SPECIFICATIONS Evolution of the design of steam generators and intermediate heat exchangers between the Phenix power plant and the design of the 1200 MW(e) power plant, 29:26126	仏	IHX SG	Na	Phenix (LMFBR)	
7	HTGR gas turbine power plant configuration studies, 29:31193 (GULF-GA-A-12651)	米			He	HTGR
20	Intermediate heat-exchanger (Na--Na) (LMFBR), 29:23513	インド	IHX			LMFBR
52	Plate heat exchangers in nuclear power stations, 29:23442	スウェーデン				
53	HEAT EXCHANGERS/STEADY FLOW Stationary and instationary flow phenomena in and behiad staggered and in-line tube banks, 29:29238 (CONF-730443-)	西独				
54	HEAT EXCHANGERS/STRESS CORROSION Fuels and Materials Development Program quarterly progress report for period ending June 30, 1973, 29:3434 (ORNL-TM-4355)	米				
25	HEAT EXCHANGERS/TEMPERATURE GRADIENTS Mean temperature difference of multichannel heat exchangers, 29:12148	ソ				
28	HEAT EXCHANGERS/TEMPERATURE NOISE Linear analytical one-dimensional model describing the frequency response behavior of shell-and-tube type counterflow heat exchangers with respect to primary and secondary inlet temperature and mass flow perturbations, 29:29223	西独				
55	HEAT EXCHANGERS/TESTING Experiences acquired from the damage to the reheater in the 50 MW test facility at Hengelo, Holland, 29:29222 (TID-26487)	オランダ	IHX SG	Na evaporator superheater reheater		LMFBR
21	HEAT EXCHANGERS/THERMAL STRESSES LMFBR problems due to large temperature differences, 29:6945	米				LMFBR
56	HEAT EXCHANGERS/TUBES Collapse of ductile heat exchanger tubes with ovality under external pressure, 29:12116 (CONF-730942-(P2))	米				

57	Experimental and analytical correlation of local driving forces and tube response in liquid flow induced vibration of heat exchangers (PHWR), 29:11818 (CONF-730942-(P2))	カ	
33	HEAT EXCHANGERS/TURBULENT FLOW Study of flow and acoustic phenomena in a tube-bank, 29:31176 (CONF-730465-)	仏	
58	HEAT EXCHANGERS/TWO-PHASE FLOW Improvements in or relating to heat exchangers (Patent), 29:11663		
53	HEAT EXCHANGERS/UNSTEADY FLOW Stationary and instationary flow phenomena in and behind staggered and in-line tube banks, 29:29238 (CONF-730465-)	西独	
59	HEAT EXCHANGERS/WELDED JOINTS Cracking in welded joints. Occurrence and prevention in nuclear plant. II, 29:20101		Na

1974 Vol. 30

74000	HEAT EXCHANGERS				
61	Helical coil heat exchanger (Patent; LMFBR), 30:8706				LMFBR
62	HEAT EXCHANGERS/CORROSION Application of eddy currents to detect corrosion in copper and stainless steel tubing, 30:18128 (K-1860)	米			
63	HEAT EXCHANGERS/DESIGN Assessment of actual loading conditions of steam generators and heat exchangers according to probabilistic criteria, 30:8838 (CONF-730942-P5)	西独	SG		
64	Augmentation of heat transfer in tubes by use of mesh and brush inserts, 30:18546	西独			
65	Flow distribution for heat exchangers (Patent), 30:29275				
66	Heat exchanger (Patent), 30:250				
67	Heat exchangers for high-temperature gas cooled reactors, 30:1881	日		He	HTGR
68	Improvements in tubular heat exchangers (Patent), 30:249				
69	Industrial technique (patent), 30:29277				
70	New construction for heat exchangers and steam generators, 30:31899 (AED-CONF-73-223-001)	西独			
71	Problems of heat dissipation in large condensation block units, 30:5608	西独			
72	Trends in structural mechanics. A UK review of the 2nd international conference on structural mechanics in reactor technology. XII. Structural analysis of heat exchangers, 30:34377	英			
73	HEAT EXCHANGERS/HEAT RESISTING ALLOYS Present situation of refractory metals and alloys for use in nuclear energy. II. Present situation and future prospect of refractory metals and alloys (Use for HTGCRs and heat exchangers), 30:7285	日		He	HTGR
74	HEAT EXCHANGERS/HEAT TRANSFER Computer codes for the calculation of the frequency response and time behavior of shell-and-tube type counterflow heat exchangers with respect to primary and secondary inlet temperature and mass flow perturbations (TUMHEX in combination with FRETI), 30:8843 (CONF-730942-P6A-S)	西独			
75	Effect of a homogeneous roughness of the surface on the hydraulic resistance and heat transfer in a turbulent air flow (GCFR), 30:23132	〃			GCFR
76	Investigation of heat and mass transfer of a vertical staggered bundle of tubes in a dispersed gas/liquid flow, 30:3060	〃			
77	Thermal and hydraulic characteristics of bundles of tubes with transverse fins in a longitudinal flow, 30:3061	〃			
76	HEAT EXCHANGERS/MASS TRANSFER Investigation of heat and mass transfer of a vertical staggered bundle of tubes in a dispersed gas/liquid flow, 30:3060	〃			
78	HEAT EXCHANGERS/MATHEMATICAL MODELS Simple dynamic models for double-pipe and tube-bundle heat exchangers, 30:23918	西独			
79	HEAT EXCHANGERS/MEETINGS Proceedings of international symposium of vibration problems in industry, Keswick, England, 10--12 April 1973. Session 4, 30:34367 (CONF-730465-(P4))				

74	HEAT EXCHANGERS/PERFORMANCE Computer codes for the calculation of the frequency response and time behavior of shell-and-tube type counterflow heat exchangers with respect to primary and secondary inlet temperature and mass flow perturbations (TUMHEX in combination with FRETI), 30:8843 (CONF-730942-P6A-S)	西独			
80	Experimental determination of the Nusselt number for the case of an exponentially increasing wall heat flux in the flow direction, 30:15323 (AREAEE-185)	エジプト			
81	Heat exchangers with orthogonal circulation in smooth tubes. Calculation of the output temperatures of the fluids, 30:236	仏			
82	HEAT EXCHANGERS/PERFORMANCE TESTING Mixed convection in the shell of a horizontal Na-to-NaK heat exchanger, 30:21004	コーラトム			Na-NaK
71	HEAT EXCHANGERS/SPECIFICATIONS Problems of heat dissipation in large condensation block units, 30:5608	西独			
83	HEAT EXCHANGERS/STANDARDS Intermediate heat exchanger for liquid metal systems, 30:14597 (RDT-E-4-6-T(5-74))	米	IHX	液体金属	
84	HEAT EXCHANGERS/STRESS ANALYSIS Assessment of actual loading conditions of steam generators and heat exchangers according to probabilistic criteria, 30:8875	オーストリア			
72	HEAT EXCHANGERS/STRESSES Trends in structural mechanics. A UK review of the 2nd international conference on structural mechanics in reactor technology. XII. Structural analysis of heat exchangers, 30:34377	英			
86	HEAT EXCHANGERS/THERMAL ANALYSIS Analysis of nonsymmetric double pass heat exchangers with nonequal overall coefficients of heat transfer, 30:15325 (AREAEE-189)	エジプト			
87	HEAT EXCHANGERS/TRANSIENTS Transient behaviour of nuclear steam generators, 30:14592 (INIS-mf-1323)	オランダ	evaporator SG	Na 水	LMFBR PWR
88	HEAT EXCHANGERS/TUBES Calculation of natural frequencies of heat exchanger tubes, 30:34368 (CONF-730465-(P4))	カ			
89	Vibration frequencies and modal shapes for multi-span beams with uniformly spaced supports, 30:3044 (CONF-740330-1)	カ			

1975 Vol. 31

75000	HEAT EXCHANGERS				
1	LMFBR intermediate heat exchanger (Patent), 31:28022	米	IHX	Na	LMFBR
2	Nuclear reactor with integrated heat exchanger (Patent: LMFBR), 31:13253			Na	LMFBR
3	HEAT EXCHANGERS/BYPASSES Industrial technique (Patent), 31:5325				
4	HEAT EXCHANGERS/CLEANING HEDL sodium removal program, annual report FY-1974, 31:32812 (HEDL-TME-74-63)	米		Na	
5	HEAT EXCHANGERS/DESIGN Heat exchanger having a head with an integral radiation shield (Patent), 31:19514	米			
6	Improvements in or relating to tube-in-shell heat exchangers (Patent), 31:8527	英			
7	Improvements in plate heat-exchangers and energy generating thermal installations comprising same (Patent), 31:16387	仏			
8	Improvements in plate heat-exchangers and energy generating thermal installations comprising same (Patent), 31:25731	仏			
9	Liquid metal operated heat exchanger (Patent), 31:32829	米	evaporator superheater	液体金属	
10	HEAT EXCHANGERS/EFFICIENCY Causes of poor heat transfer in shell-and-tube heat exchangers during condensation of saturated steam, 31:11096 (AERE-Trans-1132)	西独			
11	HEAT EXCHANGERS/EXPLOSION WELDING Inspection and explosive weld plugging of defective tubing in a ferritic heat exchanger, 31:35608				
12	HEAT EXCHANGERS/FAILURES Operating experience with heat exchangers in nuclear power stations with light water reactors, 31:35600	西独	SG	水	Beznau 1,2 (LWR)
13	HEAT EXCHANGERS/FLUID FLOW Flows in heat exchanger plenums, 31:22536 (ANL-CT-76-05)	米			
14	HEAT EXCHANGERS/HEAT TRANSFER Core auxiliary heat exchanger for a high-temperature gas-cooled reactor: water side static stability analysis and design, 31:3177	米		He	HTGR
15	CSDT simulation midel for prediction of IHX transient behavior using S/360 CSMP (LMFBR; CSMP code), 31:21766 (ANL-8118)	米	IHX	Na	LMFBR
16	Radiation effects on heat transfer in reactor core and heat exchangers of HTGR, 31:31775	日		He	HTGR
11	HEAT EXCHANGERS/INSPECTION Inspection and explosive weld plugging of defective tubing in a ferritic heat exchanger, 31:35608				
18	HEAT EXCHANGERS/LEAK TESTING Refinery heat exchanger leak test using krypton-85, 31:14257 (INS-R-121)	ニュージーランド			
19	HEAT EXCHANGERS/MECHANICAL VIBRATIONS Use of the PAS-1 finite element computer system for static and dynamic analysis of two- and three-dimensional structures (PWR heat exchanger tubes), 31:24614	伊		水	PWR
20	HEAT EXCHANGERS/NONDESTRUCTIVE TESTING Automatic testing systems development brought about by nuclear construction, 31:475	仏			

21	HEAT EXCHANGERS/PERFORMANCE HTGR safety studies progress report for the period ending March 31, 1974, 31:2516 (ORNL-TM-4736)	米		He	HTGR
22	HTGR safety studies quarterly progress report for the period ending June 30, 1974, 31:2517 (ORNL-TM-4737)	米		He	HTGR
23	New ways of predicting heat-transfer performance (Computer codes), 31:29520	英			
24	HEAT EXCHANGERS/PLATES Thermal stresses on the surface of perforated plates with temperature distributions varying through the thickness only, 31:19472 (WAPD-TM-1190)	米			
25	HEAT EXCHANGERS/SIMULATION Simulation techniques of distributed parameter systems using a hybrid computer, 31:21967 (JAERI-M-5733)	日			
26	HEAT EXCHANGERS/SPECIFICATIONS Assessment of actual loading conditions of steam generators and heat exchangers according to probabilistic criteria, 31:5321	オーストリアSG			
14	Core auxiliary heat exchanger for a high-temperature gas-cooled reactor: water side static stability analysis and deisgn, 31:31777	米		He	HTGR
19	HEAT EXCHANGERS/STRESS ANALYSIS Use of the PAS-1 finite element computer system for static and dyanmic analysis of two- and three-dimensional structures (PWR heat exchanger tubes), 31:24614	伊		水	PWR
20	HEAT EXCHANGERS/SUPPORTS Industrial technique (Patent), 31:5323				
27	HEAT EXCHANGERS/TEMPERATURE CONTROL A method and device for regulating the temperature of a heat exchanger primary fluid (Submarine thermionic reactors; patent), 31:32181				サブマリン
28	HEAT EXCHANGERS/THERMAL STRESSES Protection device for tube-plate heat-exchangers (Patent), 31:16481				
15	HEAT EXCHANGERS/TRANSIENTS CSDT simulation model for prediction of IHX transient behavior using S/360 CSMP (LMFBR; CSMP code), 31:21766 (ANL-8118)	米	IHX	Na	LMFBR
29	HEAT EXCHANGERS/TUBES 2-1/4-percent-chromium, 1-percent-molybdenum alloy steel seamless tubes (ASME SA-213 with additional requirements), 31:30169 (RDT-M-3-33T(2-75))	米			
30	HEAT EXCHANGERS/WELDING Studies on internal welding of small-size tubes (Tube/tube plate connections), 31:11887	日			

1975 Vol. 23

75000	HEAT EXCHANGERS See also RADIATORS				
31	A heat-exchanger apparatus, specially suitable as steam generator (Patent), 32:16625				
32	Investigations on HTR-heat exchanger materials, 32:2416 (JUL-1111-RG)	西独		He	HTR
33	Nuclear reactor (GCR type reactors Patent), 32:16252				GCR
34	HEAT EXCHANGERS/CONVECTION Calculation of convective heat transfer in complex flow systems, 32:19653	英	coolingtower condenser		
35	HEAT EXCHANGERS/DESIGN Design guide for heat transfer equipment in water-cooled nuclear reactor systems, 32:18802 (ORNL-TM-3578)	米		水	LWR
36	Heat exchangers: design and theory sourcebook, 32:439				
37	Heat transfer design and proof tests of a radioisotope thermoelectric generator. Final report, November 1972--July 1974 (For underwater use), 32:3288 (AD/A-002218)	米			
38	Improvements in or relating to heat exchangers (Patent), 32:3306				
39	Liquid metal operated heat exchanger (Patent), 32:8765			液体金属	
40	Nuclear engineering power plant component, 32:24001	英			
41	Some extra-high capacity heat exchangers of special design, 32:440	ルーマニア			
42	HEAT EXCHANGERS/FORCED CONVECTION Forced convection heat transfer in the transition from laminar to turbulent flow in closely spaced circular tube bundles, 32:19612	西独			
43	HEAT EXCHANGERS/HEAT TRANSFER Analytical treatment of two-dimensional heat conductivity in a rectangular cross section under given boundary conditions (HTGR heat exchangers), 32:10756 (JUL-1172)	西独		He	HTGR
44	Cooling system for discharge or process heat into the air without evaporation, 32:27897 (PATENT)	西独			
45	Preliminary study on high temperature heat exchanger for nuclear steel making, 32:18841	日		He	HTGR
46	HEAT EXCHANGERS/LIQUID FLOW Developing laminar radial velocity profiles and pressure drop in the entrance region of concentric annuli, 32:5704	英			
47	HEAT EXCHANGERS/MATHEMATICAL MODELS Analytical one-dimensional frequency model for U-tube heat exchangers with respect to the input and output mass flow temperatures on the primary and secondary sides, 32:21719 (MRR-146)	西独			
48	HEAT EXCHANGERS/MECHANICAL VIBRATIONS Investigation of vibration of a cylinder by parallel liquid flow, 32:19162	日			
49	Vibrations of a row of circular cylinders in a liquid (LMFBR component vibration studies), 32:10771 (ANL-CT-75-34)	米			LMFBR
50	HEAT EXCHANGERS/MOCKUP 50 MWe test loop. Monthly report, June 1974 (weeks 23--26) (Sodium coded reactors), 32:30412 (TNO-LMEC-6)	オランダ	IHX,SG evaporator	Na	
51	50 MWe test loop. Monthly report, September 1974 (weeks 36--39) (Sodium cooled reactors), 32:30414 (TNO-LMEC-8)	オランダ	IHX,SG	Na	
52	Department for operation of 50 MW test loop. Monthly report, May 1974 (weeks 18--22) (Sodium cooled reactor mockup), 32:5697 (TNO-LMEC-5)	オランダ	IHX,SG evaporator	Na	

53	HEAT EXCHANGERS/PERFORMANCE Aspects of heat transfer in two-phase gas--liquid flow, 32:610	仏		
54	Utilization of fluid temperature fluctuations to assess the thermal performance of heat exchange equipment (Gas flow past complex heat transfer surfaces), 32:19597 (CONF-730578-)	オーストラリア		
50	HEAT EXCHANGERS/PERFORMANCE TESTING 50 MWe test loop. Monthly report, June 1974 (weeks 23--26) (Sodium coded reactors), 32:30412 (TNO-LMEC-6)	オランダ	IHX,SG evaporator	Na
51	50 MWe test loop. Monthly report, September 1974 (weeks 36--39) (Sodium cooled reactors), 32:30414 (TNO-LMEC-8)	オランダ	IHX,SG	Na
52	Department for operation of 50 MW test loop. Monthly report, May 1974 (weeks 18--22) (Sodium cooled reactor mockup), 32:5697 (TNO-LMEC-5)	オランダ	IHX,SG evaporator	Na
55	Heat transfer design and proof tests of a radioisotope thermoelectric generator. Final report, November 1972--July 1974 (For underwater use), 32:3288 (AD/A-002218)	米		
56	HEAT EXCHANGERS/THERMAL STRESSES Evaluation and thermal shock test of weldable strain gage in hot sodium environment (560°C), 32:19834 (ORNL-tr-2998)	日	IHX	Na
57	Thermal effects on tubesheets, 32:16613 (EP-74-R-34)	カ		
58	HEAT EXCHANGERS/TURBULENT FLOW SS-TVF a computer code for turbulent flow velocity field calculations in square array rod subchannels, 32:19168 (RT/ING-(74)29)	伊	SG	
59	TS-TVF a computer code for turbulent flow velocity field calculation in triangular array rod subchannels, 32:19169 (RT/ING-(74)28)	伊	SG	
60	HEAT EXCHANGERS/TWO-PHASE FLOW Aspects of two-phase gas--liquid flow, 32:611	英		
61	HEAT EXCHANGERS/WELDED JOINTS Inspection and explosive weld plugging of defective tubing in a ferritic heat exchanger, 32:6353 (AED-CONF-74-538-051)	英		

1976 Vol. 33

76000	HEAT EXCHANGERS See also RADIATORS				
1	Changes in the conception of the steam generators and the intermediate exchangers between the Phenix plant and the 1200 MWe plant (LMFBR), 33:16285 (ANL-Trans-1009)	仏	IHX,SG	Na	Phenix (LMFBR)
2	Energy absorber for sodium-heated heat exchanger (Patent), 33:28058				
3	Tube heat exchanger (LMFBR type reactors; patent), 33:14018				
4	HEAT EXCHANGERS/COST Heat exchangers for high-temperature thermodynamic cycles (Fossil-fueled and nuclear power plants), 33:399 (CONF-751106-4)	米	IHX,SG		
5	HEAT EXCHANGERS/CROSSFLOW SYSTEMS Literature search of high velocity cross flow fluid, 33:20153 (CNLM-6351)	米			
6	HEAT EXCHANGERS/DECONTAMINATION PWR radiation buildup: the course ahead, 33:22004	米		水	PWR
7	HEAT EXCHANGERS/DESIGN Development of a heat source assembly for an isotope Brayton space power conversion system, 33:5669	米			ブレイトンスペース
8	^{4}He evaporation heat exchangers for a ^{3}He/^{4}He dilution refrigerator, 33:5504	日		He	
4	Heat exchangers for high-temperature thermodynamic cycles (Fossil-fueled and nuclear power plants), 33:399 (CONF-751106-4)	米			
9	Heat exchanger (Patent), 33:22579				
10	Helium dilution refrigerator ^{3}He--^{4}He verdunningskoelmachine, 33:14761 (P)				
11	Hydraulic problems in sodium exchangers and repercussion in thermal areas (Noniniform flow), 33:4672 (ANL-Trans-1006)	仏		Na	Phenix (FBR)
12	Hydraulic design considerations for a multi-tube sodium economizer, 33:22183	米		Na	
13	Hydraulic design considerations for a multi-tube sodium economizer, 33:28820 (HEDL-SA-926)	米		Na	
14	HEAT EXCHANGERS/EDDY CURRENT TESTING Adaptation of eddy current methods to the multiple problems of reactor testing, 33:17134 (AED-CONF-75-179-004)	西独			
11	HEAT EXCHANGERS/EFFICIENCY Hydraulic problems in sodium exchangers and repercussion in thermal areas (Noniniform flow), 33:4672 (ANL-Trans-1006)	仏		Na	
15	HEAT EXCHANGERS/FABRICATION Application of electron beam welding to the joining of a thin-wall ''J'' probe to a thin-wall adapter, 33:20717 (CNLM-5756)	米			
7	Development of a heat source assembly for an isotope Brayton space power conversion system, 33:5669	米			ブレイトンスペース
16	Development forming techniques to fabricate heat exchanger jackets: T-1025443, T-1025444, 1015782, and 1015786. Interim report, 33:20712 (CNLM-3529)	米			
17	HEAT EXCHANGERS/FAILURES Consequences of improper evaluation of corrosion related failures: case histories reviewed (AISI 440C cracking by overload; heat exchanger tube failure; Type 304 ss bellows stress corrosion), 33:7766 (KAPL-P-4042)	米			

13	HEAT EXCHANGERS/FLUID FLOW Hydraulic design considerations for a multi-tube sodium economizer, 33:28820 (HEDL-SA-926)	米		Na	
18	Waterside flow problems in boilers, 33:6689	米	SG		
19	HEAT EXCHANGERS/FUNCTIONAL MODELS The use of models to estimate the performance of boilers for gas cooled reactors, 33:28008	英		He	GCR
20	HEAT EXCHANGERS/GAS FLOW Analytical calculation of regenerative counterflow heat exchangers, 33:6869	ソ	RHX		
21	Method for the calculation of the gaseous counterflow regenerator for chemically reacting coolants, 33:6709	ソ		N_2O_4	
22	Solution of some problems of thermophysics by approximation method, 33:14769 (AEC-tr-7295)	ソ			
23	HEAT EXCHANGERS/HEAT TRANSFER Calculation of the parameters of chemically reacting flow through a heated channel (N_2O_4 flow), 33:14768 (AEC-tr-7295)	ソ		N_2O_4	
24	Heat transfer to finned tubes and computation of tube bank heat exchangers, 33:534 (BNWL-tr-155)	西独			
25	Heat transfer in turbulent separated flow, 33:29036	英			
26	Radiation effects on heat transfer in the reactor core and heat exchangers of an HTGR, 33:6707	日		He	HTGR
27	HEAT EXCHANGERS/HYDRAULICS Study on hydraulic characteristics of tubular space of heat-exchangers, 33:32466	ソ			
28	HEAT EXCHANGERS/INSPECTION Advanced absorber materials (Eu_2O_3 and its cermets; weldments for LMFBRs; post-test examination of SCTI-IHX), 33:3404 (ORNL-TM-5029)	米	IHX	Na	LMFBR
29	HEAT EXCHANGERS/LEAK TESTING Helium leak testing of the heat exchangers of the Bugey-1 nuclear power plant, 33:28011	仏		He	Bugey-1 (GCR)
30	Method and apparatus for testing closed-end tubes in heat exchangers of nuclear reactors and the like (Patent), 33:19597				
31	HEAT EXCHANGERS/LEVELS Liquid level controller (Patent), 33:11277				
32	HEAT EXCHANGERS/MECHANICAL PROPERTIES Creep-rupture properties of a weld-overlaid type 304 stainless steel forging (482, 538, 593°C; FFTF IHX tubesheet), 33:23735 (ORNL-5085)	米	IHX	Na	FFTF
33	HEAT EXCHANGERS/MECHANICAL VIBRATIONS Dynamics of tube-baffle interaction in heat exchangers, 33:22559	米			
34	Experimental studies of tube/baffle interaction in heat exchangers, 33:22558	米			
35	HEAT EXCHANGERS/OPERATION Heat exchanger analysis in the case of incomplete information, 33:14612				
36	HEAT EXCHANGERS/PERFORMANCE Thermal transport properties of helium, helium--air mixtures, water, and tubing steel used in the CACHE program to compute HTGR auxiliary heat exchanger performance, 33:25144 (ORNL/TM-4931)	米		He He+air 水	HTGR
7	HEAT EXCHANGERS/PERFORMANCE TESTING Development of a heat source assembly for an isotope Brayton space power conversion system, 33:5669	米			ブレイトン
37	Experiments with miniature heat exchangers for dilution refrigerators, 33:5505				

38	Experimental test-bed Di-P for testing models of heat exchangers (Dissociating N_2O_4), 33:14763 (AEC-tr-7295)	ソ			
39	Industrial demonstration trials, 33:29480	仏			
40	Some trials for novel heat exchangers for a dilution refrigerator, 33:5508	日			
41	HEAT EXCHANGERS/RESEARCH PROGRAMS Bubble behavior on the scale-up fluidized bed heat exchangers. Quarterly technical progress letter, 1 September 1975--30 November 1975, 33:11785 (PERC-0055-1)	米			
42	HEAT EXCHANGERS/REVIEWS Heat exchangers. Fundamentals (Lecture proceedings), 33:23236	仏			
43	HEAT EXCHANGERS/SHIELDS Circulating equipment for radioactive liquid (Patent), 33:2851				
44	HEAT EXCHANGERS/SIMULATION Code of hybrid calculation for the study of heat exchangers, 33:28067	仏			
45	Digital and hybrid simulation of heat exchangers for thermal or nuclear power plants, 33:28194	仏		CO_2 Na	GCR FBR
46	Simulation of a boiler in transient operating conditions on a digital computer (Carbon diloxide cooled reactors), 33:28010	仏		CO_2	GCR
47	HEAT EXCHANGERS/SPECIFICATIONS HE-12CL test heat exchanger (Aircraft propulsion reactors), 33:19520 (CNLM-2644)	米			aircraft propulsion
4	Heat exchangers for high-temperature thermodynamic cycles (Fossil-fueled and nuclear power plants), 33:399 (CONF-751106-4)	米	IHX,SG		
48	Rational layouts of straight-tube heat exchangers for nuclear station reheaters, 33:22574	ソ	reheater		
20	HEAT EXCHANGERS/TEMPERATURE GRADIENTS Analytical calculation of regenerative counterflow heat exchangers, 33:6869	ソ	RHX		
49	HEAT EXCHANGERS/TESTING Ten years synthesis of heat exchangers testing. List of controlled performances (Power reactors), 33:28009	仏		GCR	
50	HEAT EXCHANGERS/THERMAL EXPANSION Expansion lyre-shaped tube (Patent), 33:11920				
4	HEAT EXCHANGERS/THERMAL STRESSES Heat exchangers for high-temperature thermodynamic cycles (Fossil-fueled and nuclear power plants), 33:399 (CONF-751106-4	米	IHX,SG		
50	HEAT EXCHANGERS/TUBES Expansion lyre-shaped tube (Patent), 33:11920				
51	Spacer supporting device for bundles of parallel tubes (LMFBR: Patent), 33:14003				
52	HEAT EXCHANGERS/ULTRALOW TEMPERATURE Heat transfer to helium. Application to the design of heat exchangers, 33:25864	仏		He	
53	HEAT EXCHANGERS/ULTRASONIC TESTING Ultrasonic inspection recess in heat exchanger and nuclear steam generator tubesheets (Patent), 33:29044				
54	HEAT EXCHANGERS/WELDED JOINTS Non-destructive testing tube-to-tubeplate welds (Sodium/water heat exchangers), 33:5561	英		Na	LMFBR

Boilers 1967 Vol. 21

67000	BOILERS see also Superheaters				
1	analysis of critical boiling in steam, effect of pulsating cycles on, 21:22097	ソ			
2	costs of steam generators for large power reactors, 21:13946(R) (ANL-7279)	米			
3	design analysis of Fort St. Vrain Reactor, 21:38447(R) (GA-6830)	米		He	Fort St. Vrain (HTGR)
4	design and fabrication of sodium-heated, 21:8495(R) (BAW-1280-40)	米		Na	
5	design and fabrication of sodium-heated, 21:30568(R) (BAW-1280-43)	米	superheater	Na	
6	design and fabrication of sodium-heated, 21:30569(R) (BAW-1280-44)	米		Na	
7	design and performance for organic Rankine cycle space power systems, 21:36236(R) (SAN-651-33)	米			ランキンサイクル
8	design and performance of AVR, 21:12173	西独		He	AVR
9	design and testing Rankine-cycle, using potassium as working fluid, 21:41104	米		K	
10	design for advanced converter reactor, comparison of parameters for radial flow and axial flow, 21:38454 (GAMD-7867)	米			
11	design for atomic power plants, 21:1516	ハンガリ			
12	design for gas-cooled reactor, 21:46997(P)				GCR
13	design for sodium-heated 500 MW and 1000 MW, 21:40528 (ANL-7290, pp 139-69)	米		Na	FBR
14	design for steam generation in boiling water reactors, 21:43124(P)				BWR
15	design for use with steam-cooled reactors, 21:21690(P)				
16	design modifications to Fermi Fast Breeder Reactor, 21:30097(R) (APDA-CFE-6)	米		Na	Fermi(FBR)
17	design of high-temperature high-pressure superheating, 21:27838	米	superheater		
18	design of integral, for reactors, 21:23682(P)				
19	design of liquid metal, for steam generation and superheating, 21:19828(P)			液体金属	
20	design of lithium--potassium, 21:30239 (NP-16767)	米			
21	design of MOSEL, calculations for, 21:27685 (JUL-422-RG)	西独			
22	design of nuclear power plant, standards for, 21:34086 (EUR-2757.e, pp 21-)	仏			HTGR
23	design of once-through, for pressurized-water reactors, 21:42747				PWR
24	design of once-through, optimum use of heat transfer data in, 21:41127	英			
25	design of single pass isopropylbiphenyl, for organic Rankine cycle power systems, 21:36235(R) (SAN-650-5)	米			ランキンサイクル
26	design of single-phase, using monoisopropyl biphenyl coolant in Rankine cycle space power systems, 21:36234(R) (SAN-650-4)	米			ランキンサイクル
27	design of sodium heated, for Very Large Fast Breeder Reactor, 21:35592 (WCAP-2872)	米		Na	FBR

28	design of steam, for use with liquid metal heating fluid, 21:24149(P)			液体金属	
29	design of supercritical, once-through, sodium heated, 21:26048 (NAA-SR-8508)	米		Na	
30	design of tubular, for steam generation and superheating in gas-cooled nuclear reactors, 21:19827(P)				GCR
31	design of vertical-tube once-through, for water reactors, 21:43190(P)				LWR
32	design of, for 30MW sodium-cooled reactors, 21:15640 (APAE-112(Vol.1))	米		Na	
33	design of, for isopropylbiphenyl, 21:10538(R) (SAN-650-3)	米			
34	design optimization for gas-cooled reactor, 21:4649	スイス			GCR
35	design parameters for 10,000 MW plant, 21:40553 (GA-6881)	米			
36	development for Fort St. Vrain Power Reactor, 21:27670(R) (GA-7086)	米		He	Fort St. Vrain (HTGR)
37	development for SNAP systems, Mercury-Rankine Program for, 21:14043(R) (NAA-SR-12256(Vol.1))	米			SNAP
38	development of liquid metal, for use with SNAP-2, 21:10503 (NAA-SR-12233)	米			SNAP-2
39	development of potassium, for SNAP-50/SPUR, 21:19820	米			SNAP-50/ SPUR
40	development of sodium-heated once-through, 21:19379 (ORNL-3955(Vol.2))	米		Na	
41	dynamic testing of Pathfinder, 21:40615	米	superheater		BWR
13	economics of sodium-heated 500 MW and 1000 MW, 21:40528 (ANL-7290, pp 139-69)	米		Na	FBR
29	economics of sodium-heated supercritical-steam, 21:26048 (NAA-SR-8508)	米		Na	
42	element welding for, for nuclear power stations, 21:12166	英		CO_2	Dungeness A
43	element welding for, for nuclear power stations, 21:12165	英		CO_2	Bradwell
44	evaluation of SNAP-8, 21:20473(R) (NASA-CR-72156)	米			SNAP-8
45	feasibility study of bellows-free secondary sodium piping for SCTI, 21:226 (NAA-SR-Memo-11941)	米			
46	feed pump for, design of, 21:15683	伊			
47	feed water purification for, using oxide-covered iron pellet beds, 21:39098(P)				
48	feedwater control system for, 21:12565(P)				
49	finned tube fabrication for, 21:28734	英		CO_2	Magnox, AGR
5	flaw detection in superheater riser tubes in sodium-heated, 21:30568(R) (BAW-1280-43)	米	superheater	Na	
50	flaw detection in welded seams in, 21:4720	ソ			
51	flow distribution to, measurement of, 21:27671 (GA-7638)	米		He	HTGR
29	fouling factors for sodium--water, 21:26048 (NAA-SR-8508)	米		Na	
52	heat exchanger for, design of liquid metal heated, 21:36257(P)			液体金属	
53	heat transfer and pressure drop in liquid metal, for use in potassium vapor Rankine system, 21:4659 (NASA-SP-131, pp 201-38)	米		液体金属	ランキン
54	heat transfer in evaporator sections of gas-cooled reactor, effects of nucleate boiling on, 21:41128	英			GCR

55	heat transfer in mercury, effect of surface oxidation on, 21:19843(R) (SAN-677-6)	米	液体金属	ランキン
56	heat transfer in mercury, for use in Rankine power cycles, 21:12575(R) (SAN-677-3)	米	液体金属	ランキン
57	heat transfer in pressurized-water, mass flow kinetics of, 21:41131	英		
58	heat transfer in steam generating, effects of surface deposits on, 21:41131			
59	installation and testing of, in Sodium Component Test Installation, 21:32293(R) (NAA-SR-12416)	米	Na	SCTI
60	kinetics of sodium-heated, analog simulation of, 21:36781 (BNL-50054)	米	Na	
61	leak testing of Shippingport, 21:40580(R) (WAPD-MRP-121)	米	水	Shipping port(PWR)
62	materials evaluation for 100-MW(e) sodium-heated, 21:452 (BAW-1280-37)	米	Na	
63	materials for, review on, 21:20495	米		
64	operation of steam. use of highly mineralized seawater in, (T), 21:43098(T) (AEC-tr-6893)	ソ		
65	performance evaluation program for sodium-cooled reactors, 21:2830(R) (UNC-5162)	米	Na	
66	performance of mercury, for use in SNAP-8, 21:20472(R) (NASA-CR-72146)	米		SNAP-8
67	performance of Peach Bottom Power Reactor, 21:25624(R) (GA-7595)	米	He	Peach Bottom (HTGR)
68	performance of rotating, for Rankine-cycle power units, 21:41110	米		ランキンサイクル
29	performance of sodium-heated supercritical-steam, 21:26048 (NAA-SR-8508)	米	Na	
69	performance of tube-in-shell mercury, measurement of heat transfer for SNAP-8, 21:44926 (NASA-TM-X-1441)	米		SNAP-8
70	power density of integral, for high-temperature gas-cooled reactors, 21:22094	オーストリア	Gas	HTGR
71	sodium-heated, analog simulation for design of 30 MW(e), 21:19790(R) (BAW-1280-42)	米	Na	
72	sodium-water reaction in sodium-heated, 21:32288 (AI-66-152)	米	Na	
73	stability and performance of, for organic Rankine cycle power systems, 21:19451(R) (SAN-651-16)	米		ランキンサイクル
74	stability and vibration testing of Fort St. Vrain Power Reactor, 21:38451(R) (GA-7314)	米	He	Fort St. Vrain (HTGR)
3	stability of Fort St. Vrain Reactor, steam/water flow, 21:38447(R) (GA-6830)	米	水－蒸気	Fort St. Vrain (HTGR)
75	stability tests of Fort St. Vrain Reactor, steam/water flow, 21:38449(R) (GA-6950)	米	水－蒸気	Fort St. Vrain (HTGR)
76	steady partial load data for once-through, use of computer to calculate, 21:34402	西独		GCR
77	steam generating tubes of, effects of vapor distribution and volume on pulsating flows in, 21:43172	ソ		
78	strain gages for sodium-heated, use of high temperature, 21:8638 (BAW-1280-41)	米	Na	
79	temperature distribution in tubes of, 21:41123			
80	testing of, 21:14127(R) (NP-16420)	米		
81	thermal stresses in tube sheets for steam, analysis of, 21:3210 (KAPL-M-6585)	米		
82	tubes for, eddy current nondestructive testing of large stainless steel, 21:19864 (BNWL-SA-1142)	米		
83	valve design for high-pressure, 21:23749	伊		
84	void levels for Sioux Falls Power Reactor, 21:38469(R) (TID-23919)	米	水	Sioux Falls (BWR)

1968 Vol. 22

68000	BOILERS see also Superheaters				
1	configuration of BN-350 Reactor, description of, 22:53294	ソ		Na	BN-350 (FBR)
2	configuration of Hinkley Point 2 Reactor, description of, 22:46970	英		CO_2	Hinkley Point 2 (AGR)
3	configuration of multiple vertical parallel row-arranged, with inner tube bundles, 22:50878(P)				
4	configuration of multiple vertical rows of straight suspended heat units for, for power reactors, 22:48677(P)				
5	construction of AGR, description of, 22:22728	英		CO_2	AGR
1	construction of liquid sodium-cooled reactor, description of, 22:53294	ソ		Na	BN-350(FBR)
6	cooling of, economics of gas, 22:16740 (WB-KE-2004)				
7	corrosion in power plant, methods for reducing, 22:52134				
8	corrosion of chromium--molybdenum steel, by mercury at 315 to 703°C, 22:4745	米			
9	corrosion of stainless steel and chromium--molybdenum, by liquid sodium, 22:32834	米		Na	
10	corrosion testing of materials for pressurized-water reactors, analysis of, 22:32835	米			PWR
11	description for 550 MW integral unit gas cooled reactor, 22:3852	英		CO_2	AGR
12	description of 400 MW vertical U-tube, for power reactors, 22:32836	米			
13	description of BN-350 Reactor, 22:18506	ソ		Na	BN-350(FBR)
14	description of EDF-4 monotube, 22:16176	仏		CO_2	EDF-4(GCR)
15	description of indirect cycle, for power reactors, 22:37454	スイス			
16	description of once-through-forced flow, for use with nuclear superheater, 22:33113(P)				
17	description of two water-pipe, vertical, 22:16345(P)				
18	design and operation of sodium-heated modular, for Sodium Component Test Installation, 22:2285	米		Na	
19	design evaluations for sodium cooled power reactor, detailed, 22:37443 (ITN/RF-1.1.1)	伊		Na	EBR-1.2 DFR (FBR)
20	design features of Dungeness reactor units, 22:7857	英		CO_2	Dungeness (AGR)
21	design for breeder reactors, 22:10438(R) (NAA-SR-12585)	米		Na	FBR
22	design for production of vapor for heat transfer systems, 22:49001(P)				
23	design for steam-cooled reactors, 22:19022(P)			蒸気	
24	design for use in sodium systems at high temperatures, review of, 22:17100 (BNL-50089)	米		Na	
25	design of Fermi Fast Breeder Reactor, 22:48633 (APDA-307)	米		Na	Fermi(FBR)
26	design of improved, for steam power plants, 22:6519(P)				
27	design of liquid entrainment and vortex generator, for cesium and potassium Rankine cycle systems for space power plants, 22:51137 (ORNL-TM-2080)	米		液体金属	Space power
28	design of once through sodium-heated, engineering drawings for, 22:40299	米	superheater reheater	Na	

29	design of shell and tube type, for steam generation, 22:6524(P)			
30	design of sodium-heated, 22:647(R) (BNL-50077)	米	Na	
31	design of sodium-heated steam generator, functional engineering calculations for, 22:42182 (BW-67-2(App.)(Vol.13)	米	Na	
32	design of steam, for Sodium Components Test Installation, 22:206(R) (NAA-SR-12520)	米	Na	
33	design parameters for power reactor, analysis of, 22:32833	米		
34	design parameters of sodium-heated steam generator, TEMA calculations on, 22:42172 (BW-67-2(App.)(Vol.3))	米	Na	
35	design specifications for 527 MW(t) sodium-heated, with 98 MW(t) sodium-heated reheat superheater, 22:35465 (KFK-654)	西独	Na	
36	development and testing for use in organic Rankine cycle space power systems, 22:30768(R) (SAN-651-57)	米		ランキンサイクル
37	development and testing for use in Rankine power cycles, 22:45221(R) (SAN-651-65)	米		ランキンサイクル
38	development for 600 MW(e) direct cycle boiling water reactors, 22:12475(R) (EURAEC-1945)	西独	水	BWR
39	development for organic Rankine cycle space power system, (E), 22:10506(R) (SAN-651-43)	米		ランキンサイクル space power
40	development for organic Rankine space power system, (E), 22:25617(R) (SAN-651-49)	米		ランキンサイクル space power
41	development of 500 to 1000 MW(e) liquid sodium-cooled fast reactor, analysis of problems in, 22:32736(T) (ANL-Trans-588)	ソ		BN-350 (FBR)
42	development of forced flow mercury, for studies of liquid metal Rankine cycle performance, 22:47285(R) (SAN-677-29)	米	液体金属	ランキンサイクル
43	development of SNAP-2 mercury, review of, 22:35573 (TRW-690-32)	米		SNAP-2
44	development of sodium-heated, 22:8247(R) (BAW-1280-47)	米	Na	
45	development of sodium-heated, 22:16735(R) (BAW-1280-48)	米	Na	
46	development of sodium-heated, 22:51133(R) (BAW-1280-56)	米	Na	FFTF
47	development of sodium-heated steam generator, evaluation of redesign alternatives in, 22:27398(R) (BAW-1280-50)	米	Na	
48	dynamic analysis of, with steam-generating, pressure-tube, heavy-water-moderated reactor heat source, (T), 22:6048	英		HWR
49	dynamics of small, performance optimization of, 22:48976 (AEEW-R-580)	英		
50	effects of liquid sodium environment on fast reactor, 22:18503	仏	Na	FBR
51	effects of sodium--water reactions in, 22:18600	英		
25	fabrication and installation of Fermi Fast Breeder Reactor, 22:48633 (APDA-307)	米	Na	Fermi(FBR)
52	fabrication of Bohunice Power Reactor, description of, 22:24997	チェコ	CO_2	Bohunice (GCR)
53	fabrication of EDF-4 reactor monotubular, description of, 22:25025	仏	CO_2	EDF-4(GCR)
54	fabrication of liquid sodium-cooled reactor austenite tubes for, description of, 22:39794	オランダ	Na	FBR
55	fabrication of nuclear power plant, welding methods for, 22:25578	仏		

56	fabrication of once-through, for Dungeness B reactors, 22:50767	英	CO_2	Dungeness B (AGR)
57	fabrication of pressurized-water reactor, description of, 22:53310	米	水	PWR
58	failures of Peach Bottom Power Reactor, analysis of type 304 stainless steel tube, 22:32840	米	He	Peach Bottom (HTGR)
59	feedwater purification for steam, effect of oxygen on, 22:10443			
60	heat transfer deterioration in supercritical fluids in, studies of, (E/T), 22:45237 (DSR-70332-55)	米		
61	heat transfer in Rankine cycle liquid metal, measurement of steady-state and transient, 22:10486(R) (SAN-677-19)	米	液体金属	ランキンサイクル
62	heat transfer in sodium-heated steam, (E/T), 22:12482	仏	Na	
63	heater and vertical pressure vessel for, description of, 22:20957(P)			
64	inlet header steady state and thermal transient stress analysis for sodium-heated steam generator, 22:42174 (BW-67-2(App.)(Vol.5))	米	Na	
65	leak detection in liquid-sodium-heated, mass spectrographic method for, 22:30471	仏	Na	
66	leak detection of sodium-water rea-tions in sodium-heated, 22:12672 (ANL-7380, pp 75-8)	米	Na	
67	materials development and quality control in fabrication of, 22:52154	伊	水	PWR
68	nozzle loading calculations for sodium-heated steam generator, 22:42172 (BW-67-2(App.)(Vol.3))	米	Na	
69	operating characteristics of prototype liquid metal cooled reactor, 22:16206	米	液体金属	
70	operating experience with organic cooled reactor, summary of, 22:32842	米	有機材	
71	operating experience with pressurized-water reactor, summary of, 22:32839	米	水	PWR
72	operation of Fermi Fast Breeder Reactor, 22:18521	米	Na	Fermi(FBR)
73	operation of steam generators for marine reactors, thermal and hydraulic analysis of, 22:44872	ソ		マリン
74	operation of steam, use of radioactive tracer method to study, 22:14725	仏	水－水蒸気	
75	outlet header steady state and thermal transient stress analysis for sodium-heated steam generator, 22:42175 (BW-67-2(App.)(Vol.6))	米	Na	
76	performance during startup of Peach Bottom Power Reactor, Unit 1, 22:32809	米	He	Peach Bottom 1 (HTGR)
77	performance evaluation for reactors, 22:10437 (NAA-SR-12534)	米		
78	performance of Fermi Fast Breeder Reactor, review of, 22:32841	米	Na	Fermi (FBR)
79	performance of Peach Bottom Power Reactor unit 1, engineering testing of, 22:30148 (GA-8040)	米	He	Peach Bottom 1 (HTGR)
80	performance of Peach Bottom Power Reactor-1 helium-side, 22:12023(R) (GA-8153)	米	He	Peach Bottom 1 (HTGR)
81	performance of sodium-cooled steam generators under transient operating conditions, 22:44840(R) (UNC-5221)	米	Na	
82	performance of sodium-heated steam, 22:48982(R) (LMEC-68-20)	米	Na	

83	performance of steam generators for pressurized-water reactors, comparison of technical and economic, 22:42270	西独		水	PWR
84	performance of, Fortran IV digital code for evaluation of thermodynamic, 22:40305 (EUR-3948)	伊			
85	performance testing of, for sodium-cooled reactors, 22:10441(R) (UNC-5189)	米		Na	
57	quality control for pressurized-water reactor, description of, 22:53310	米		水	PWR
86	radiation dosimetry in vicinity of, during testing with γ radiometric pressure gages, 22:45579	仏			
87	reheater stress calculations for sodium-heated steam generator, 22:42173 (BW-67-2(App.)(Vol.4))	米		Na	
88	safety studies on sodium cooled reactor sodium-water, 22:53354	仏		Na	
89	sodium inlet nozzle fatigue analysis for sodium-heated steam generator, 22:42179 (BW-67-2(App.)(Vol.10))	米		Na	
90	sodium inlet nozzle steady state stress analysis for sodium-heated steam generator, 22:42180 (BW-67-2(App.)(Vol.11)	米		Na	
91	sodium inlet nozzle thermal transient stress analysis for sodium-heated steam generator, 22:42181 (BW-67-2(App.)(Vol.12))	米		Na	
92	sodium-water reactions in liquid sodium cooled fast reactor, effects of, 22:53355	ソ		Na	FBR
93	sodium-water reactions in LMFBR, effects of, 22:14129(R) (APDA-212)	米		Na	LMFBR
94	stress calculations for sodium-cooled reactor steam generator, 22:42170 (BW-67-2(App.)(Vol.1))	米		Na	
95	stress calculations for sodium-cooled reactor steam generator 22:42171 (BW-67-2(App.)(Vol.2))	米		Na	
96	superheater inlet header steady state and thermal transient stress analysis for sodium-heated steam generator, 22:42176 (BW-67-2(App.)(Vol.7))	米	superheater	Na	
97	superheater outlet header steady state stress analysis for sodium-heated steam generator, 22:42177 (BW-67-2(App.)(Vol.8))	米	superheater	Na	
98	superheater outlet header thermal transient stress analysis for sodium-heated steam generator, 22:42178 (BW-67-2(App.)(Vol.9))	米	superheater	Na	
99	support system for Dungeness B Reactor, vibration and corrosion design criteria for, 22:53297	英		CO_2	Dungeness B (AGR)
100	testing for failure and malfunction by analysis of flow noises in, 22:14607 (CONF-671011-, pp 314-39)	米			
101	testing of steam in Sodium Components Test Installation, 22:47266 (NAA-SR-Memo-11753)	米		Na	
102	tube erosion in high-temperature steam, 22:43529 (BAW-1280-53)	米		高温蒸気	
103	tube sheets for Fermi Fast Breeder Reactor, experience with leaking joints and cracks in the ligaments of, 22:48633 (APDA-307)	米		Na	Fermi (FBR)
104	tube sheets for sodium-heated steam generator, steady state stress calculations for, 22:42172 (BW-67-2(App.)(Vol.3))	米		Na	
105	tube wastage from sodium--water reactions in, 22:12970			Na	
106	tube wastage in sodium-heated steam generators from sodium-water reaction, (E), 22:47040(R) (APDA-22)	米		Na	

1969 Vol. 23

69000	BOILERS see also Superheaters			
1	analysis of scale from, for vanadium by neutron activation and γ spectrometry, 23:11458	日		
2	configuration of hollow tube-surrounding core-enclosed integral upward-coolant-flow superheating, description of, 23:2100(P)			
3	configuration of reactor intermediate controlled-flow steam and superheat forced-flow, 23:2139(P)		蒸気	
4	construction of boiling and pressurized water power reactor, problems of, 23:47149	西独	水	BWR,PWR
5	coolant gas flow in Fort St. Vrain, velocity profiles and temperature contours for, 23:4144 (GAMD-8625)	米	He	Fort St. Vrain(HTGR)
6	corrosion of tubes in KNK Power Reactor, by steam--water, 23:45437 (ANL-7520(Pt.1), pp 335-44)	西独	Na	KNK
7	corrosion problems in, 23:18180(T)(ANL-Trans-703, pp 9-44)	仏		
8	decontamination process for NPR stainless steel-tubed steam generators, 23:15551 (DUN-SA-74)	米		
9	description of horizontal coolant flow, for use with gas-cooled reactors, 23:1997(P)			GCR
10	design characteristics of THTR, general, 23:21439	西独	He	THTR
11	design concepts for 1000 MW(e) LMFBR sodium heated steam generator, technical and economic evaluation of, 23:33077 (WARD-2000-22)	米	Na	LMFBR
12	design concepts for liquid sodium cooled fast reactor, review of problems in, 23:47151 (ANL-7520(Pt.2), pp 3-19	米	Na	LMFBR
13	design for gas cooled and sodium cooled power reactors, 23:45100 (CONF-681064-4)	英	Na ガス	LMFBR GCR
14	design for improved heat transfer for use in gas-cooled reactors, 23:33314(P)			GCR
15	design for production of high-pressure steam for nuclear power plant, 23:2489(P)		高圧蒸気	
16	design for sodium-cooled fast reactors, 23:7689(P)		Na	高速炉
17	design of integrated 600 MW(e) gas cooled reactor, effects of engineering and economic factors on, 23:28481	英	CO_2	AGR
18	design of liquid sodium heated BN-350 Reactor steam, 23:52430(T) (JPRS-48330, pp 252-68)	ソ	Na	BN-350 (FBR)
19	design of once-through forced-flow, for nuclear power plants, 23:38154(P)			
20	design of pressure vessel enclosed superheating reactor, 23:40351(P)			
21	design of sodium-heated, for LMFBR systems, selection of optimum, 23:2458 (BAW-1280-55)	米	Na	LMFBR
22	design of steam generator for liquid metal cooled reactor, 23:37607(P)		液体金属	
23	design of steam, for nuclear power plants, 23:21869(P)			
24	design parameters for carbon dioxide cooled power reactor pod, 23:28482	英	CO_2	AGR
25	design parameters for high temperature gas cooled reactor, 23:37606	英	He	HTGR
26	design parameters for integral power reactor annular shaped, optimum economic, 23:43110(P)			

27	design parameters for integrated gas cooled power reactor, effects of configuration on, 23:28478	オーストリア		GCR
28	design parameters for large pressurized water reactor, 23:28394	米	水	PWR
29	design parameters for nuclear power plants, 23:38136 (CONF-680625-6)	英		
30	design parameters for pressurized water reactor, comparison of optimum, 23:31164	ポーランド	水	PWR
31	design requirements for 300 MW(e) Thorium High Temperature Reactor, 23:28479	西独	He	THTR
32	development for organic Rankine cycle power conversion system, 23:9613(R) (SAN-651-71)	米		ランキンサイクル
33	development of Atomics International and Babcock and Wilcox liquid sodium, description of, 23:11264 (CONF-680419-, pp 354-402)	米		
34	development of NERO Reactor steam generator, 23:45119(R) (RCN-97(Vol.4)(Pt.2))	オランダ superheater		NERO
35	development of optimum LMFBR-type reactor recirculating and once-through, comparison of, 23:2062	米		LMFBR
36	development of sodium heated, review on, 23:18182 (BNL-50153)	米	Na	
37	development of sodium-heated, review of, 23:41529 (BNL-50179	米	Na	
38	development of space-flight fast liquid lithium-cooled power reactor boiling potassium, description of, 23:6033	米	NaK Li	スペースフライト
39	development of steam generators for power reactors, review of, 23:19519	チェコ		
40	dynamic response to load variations of pressurized water reactor, analysis of calculated, 23:26006	伊	水	PWR
41	effects of Loeffler, on kinetics of superheating fast reactors, 23:43099	日 superheater		高速炉
42	fabrication of Dodewaard Power Reactor, during 1967, 23:19397(T) (AEC-tr-7032) EVR-4065	オランダ	水	Dodewaard (BWR)
43	fabrication of high-temperature power reactor, state-of-the-art for, 23:15603(T) (ORNL-tr-3010)	西独		HTR
44	fabrication of large, ASME Boiler and Pressure Vessel Code for quality assurance in, 23:26435	米		
45	fabrication of Obrigheim reactor, 23:15591	西独	水	Obrigheim (PWR)
46	fabrication of power reactor standardized, description of assembly-line, 23:23844			
47	failure of liquid sodium and water, testing of pressure transients in, (E), 23:11264 (CONF-680419-, pp 354-402	米	Na	Fermi (FBR)
48	failures of steam, evaluation of pressure vessel safety from, 23:43430 (AHSB(S)R-161)	英		
49	feedwater conditions for Novo Voronezh Power Reactor Unit 1, description of, 23:5980	ソ	水	Novo Voronezh (PWR)
50	flow stability of Fort St. Vrain Power Reactor Unit 1, testing of steam--water, 23:47169(R) (GA-9440)	米	He	Fort St. Vrain (HTGR)
51	gas mixing in cross-flow tube bundle of, 23:31472 (GAMD-8766)	米		

52	heat transfer and operational stability of, for organic Rankine power conversion system, 23:50027(R) (SAN-651-107)	米		ランキン
53	heat transfer and operational stability of, for organic Rankine power conversion system, 23:19396(R) (SAN-651-77)	米		ランキン
54	heat transfer in, computer program for calculation of, 23:33322 (EUR-4252(Pt.1))	伊		
55	heat transfer in, design of tube banks for optimum, 23:2512	ソ		
56	heat transfer measurement for once through organic Rankine cycle, 23:33324 (SAN-651-94)	米		ランキン
57	hydrodynamics for power reactor forced circulation, (T), 23:26044	チェコ		
6	hydrodynamics of tube-in-tube KNK Power Reactor, 23:45437 (ANL-7520(Pt.1), pp 335-44)	西独	Na	KNK
58	kinetics of once-through, mathematical model for simulation of, 23:26438 (AEEW-M-794)	英		
59	leak detection in sodium-heated, by hydrogen measurements, 23:45483 (ANL-7520(Pt.1), pp 345-73)	仏	Na	
60	materials for sodium-heated, properties of, 23:46038 (ANL-7520(Pt.1), pp 327-34)	米	Na	
61	mercury containment in SNAP-8, use of chromium--molybdenum steel for, 23:29965 (N-69-20195)	米		SNAP-8
62	modification of modular type steam, for use with sodium cooled reactors, 23:52396(T) (JPRS-48330, pp 313-24)	チェコ	Na	
63	operating characteristics of sodium-heated, 23:17643(R) (BAW-1280-57)	米	Na	
64	operating saturation conditions in, simulation of, 23:41131(R) (IDO-17299)	米		
65	performance of chromium--molybdenum steel, structural and thermal, 23:40920 (UNC-5236)	米		
66	performance of naval reactor, detailed analysis of, 23:26103			マリン
56	performance of once through Dowtherm-A, for use in space power systems, 23:33324 (SAN-651-94)	米		スペースパワー
67	performance of once-through, effects of heat transfer on, 23:2506	伊		
68	performance of sodium-heated once-through steam, mathematical model of dynamic, 23:47696 (KFK-955)	西独	Na	
69	performance of sodium-heated steam, 23:24344(R) (LMEC-69-1)	米	Na	
70	performance of sodium-heated steam, design of facility for testing, 23:9609(R) (LMEC-68-1)	米	Na	
71	performance of sodium-heated steam, design of facilities for testing, 23:9610(R) (LMEC-69-13)	米	Na	
72	performance of superheating fast breeder power reactor spray type, 23:28480	スウェーデン		FBR
73	performance testing of double containment tantalum-stainless steel, for SNAP-8, 23:15692 (N-68-31043)	米		SNAP-8
74	physics parameters for 1000 MW(e) HTGR-type reactor, updated conceptual design, 23:4166(R) (GA-7801)	米	He	HTGR
75	piping for, design of coupling to permit thermal expansion of, 23:33299(P)			
76	repairs to Fermi Fast Breeder Reactor, description of, 23:11264 (CONF-680419-, pp 354-402)	米	Na	Fermi (FBR)

77	sodium--water reactions in liquid sodium heated, mathematical model and analog simulation of, 23:19517 (CEA-CONF-1124)	仏	Na	
78	stability margins for naval reactor, determination of, 23:26103			マリン
79	testing iron-base tubes for, using steam at 600°C and 1400 psia, 23:29962 (GA-9299)	米		
80	testing of Fermi Fast Breeder Reactor No. 2, analysis of thermal shock, 23:19399 (APDA-CFE-26)	米	Na	Fermi(FBR)
81	testing of Fermi Fast Breeder Reactor No. 3, hydraulic and thermal shock, 23:32973(R) (APDA-CFE-28)	米	Na	Fermi(FBR)
82	testing of Fermi Fast Breeder Reactor, description of water tube integrity, 23:13355 (APDA-CFE-25)	米	Na	Fermi(FBR)
83	testing of Fort St. Vrain Power Reactor Unit 1, analysis of vibration, 23:9117(R) (GA-7453)	米	He	Fort St. Vrain (HTGR)
84	testing of Fort St. Vrain Power Reactor Unit 1, analysis of vibration, 23:9118(R) (GA-7634)	米	He	"
85	testing of sodium-heated, design of facility for leak, 23:45484 (ANL-7520(Pt.1), pp 374-83)	西独	Na	LMFBR
86	thermal protection of, design and performance of gamma radiometric level indicators for, 23:16168			NERO
87	thermodynamic testing of 6 MW scale model of NERO steam generator, 23:7408	オランダ		Palisades (PWR)
88	transport of Palisades Power Reactor, description of mobile heavy-lift equipment and techniques for, 23:23837	米	水	
40	tube bundles for pressurized water reactor, structural analyses and materials for, 23:26006	伊	水	PWR
89	tube vibration in steam, engineering method for evaluation of, 23:26405 (GA-8292)	米		
90	tube wastage from sodium-water reactions in, effects of leak rate on, 23:52483(R) (APDA-236)	米	Na	
91	tubes for fast breeder reactor, strength of chromium--molybdenum--steel, 23:10214	日	Na	FBR
92	tubes for LMFBR, measurements of sodium-induced wastage of, 23:28458(R) (APDA-234)	米	Na	LMFBR
93	tubes for N Reactor, performance of Inconel 600, 23:36096	米		
94	tubes for Obrigheim Power Reactor, fabrication of nickel alloy 600, 23:23831	西独	水	Obrigheim (PWR)
95	tubes in Tokai-Mura Power Reactor, mockup analysis of vibration damage to, 23:9101	日	水	
96	tubes in, description of two-phase fluid flow in vertical, 23:21878 (CONF-680118-1)	英		
97	uses of conventional, in nuclear power plants, 23:47678	西独		
98	weldments for HTGR, properties of nickel alloy, 23:8276 (ORNL-4370, pp 165-74)	米	He	HTGR

1970 Vol. 24

70000	BOILERS see also Superheaters			
1	anchor bolts for nuclear steam, stress analysis for steel, 24:9555	米		
2	behavior of Mutsu Reactor, effects of load changes and ship motion on transinet, 24:2055	日	水	マリンムツ(PWR)
3	burnout in sodium-heated steam, mathematical simulation of, 24:36501	仏	Na	LMFBR
4	construction requirements for EDF-4 Sulzer monotube, 24:22378	仏	CO_2	EDF-4(GCR)
5	corrosion of tubes of, by sodium in-leakage, 24:42049 (APDA-254)	米	Ba	
6	corrosion of Type 304 stainless steel tubes in Shippingport Power Station, by sodium hydroxide, evaluation of, 24:960	米	水	Shippingport (PWR)
7	decontamination of radiochemical processing plant, equipment and methods for, 24:19124 (DUN-5935)	米		
8	decontamination of radiochemical processing plant, equipment and methods for, 24:19125 (DUN-6075)	米		
9	design criteria for Fort St. Vrain Power Reactor, review of, 24:9549 (GA-9342)	米	He	Fort St. Vrain (HTGR)
10	design for fluid cooled power reactors, 24:47889(P)			
11	design for LMFBR modular, 24:7164	米		LMFBR
12	design for radioisotope-fueled Rankine cycle power system, 24:46436(R) (ORNL-TM-3099)	米		ランキン
13	design for radioisotope-fueled Rankine cycle power system, 24:36508(R) (ORNL-TM-3011)	米		ランキン
14	design of conceptual 1000 MW(e) LMFBR steam, 24:1964 (BAW-1328(Vol.3))	米	Na	LMFBR
15	design of Ft. St. Vrain Power Reactor Unit 1, 24:18643	米	He	Fort St. Vrain (HTGR)
16	design of high pressure saturated-steam, for steam-cooled nuclear reactors, 24:9568(P)		蒸気	
17	design of high-efficiency steam, for water-cooled reactors, 24:9559(P)		水	LWR
18	design of improved steam, for reactor power plants, 24:2350(P)			
19	design of improved steam, for reactor power plants, 24:2349(P)			
20	design of improved steam, with demountable headers, 24:25167(P)			
21	design of once-through steam, 24:2364(P)			
22	design of once-through steam, for use with gas-cooled reactor, 24:5807(P)			GCR
23	design of once-through, for Rankine cycle land-based electric power plants, 24:25199 (NASA-SP-5057, pp 35-89)	米		ランキン
24	design parameters for 1000 MW(e) reference design high temperature gas cooled reactor, 24:38490 (DP-Report-135(Pt.2))	英	He	HTGR
25	design parameters for EDF-4 Sulzer monotube, 24:22378	仏	CO_2	EDF-4(GCR)
26	design parameters of carbon dioxide cooled power reactor sectional, 24:28876	チェコ	CO_2	GCR
27	design parameters of Fort St. Vrain Power Reactor Unit 1, 24:9063	米	He	Fort St. Vrain(HTGR)

28	design parameters of gas cooled reactor pressure vessel mounted pod type, 24:28882	英	CO_2	AGR(GCR)
29	design parameters of Hartlepool Power Reactors, 24:9054	英	CO_2	Hartlepool (AGR)
30	design parameters of liquid sodium heated steam, 24:52899	英	Na	LMFBR
31	design, fabrication, and materials selection for sodium-cooled fast breeder-reactor, 24:1029	日	Na	LMFBR
32	development in France for liquid metal heated steam, 24:52898	仏	液体金属	
33	development in Japan of liquid sodium heated steam, 24:52897	日	Na	
34	development in Netherlands for liquid metal cooled fast reactors, 24:52896	オランダ	液体金属	高速炉
35	development in U.K. for gas cooled power reactors, 24:28877	英	CO_2	GCR
36	development of liquid sodium cooled fast reactor, CNEN program for, 24:16226	伊	Na	LMFBR
37	development of liquid sodium cooled reactor, effects of sodium--water reactions on, (E), 24:43730 (APDA-257)	米	Na	
38	development of liquid sodium heated steam, 24:52901	米	Na	LMFBR
30	development of liquid sodium heated steam, 24:52899	英	Na	LMFBR
39	development of liquid sodium heated, in France for fast reactors, 24:52909	仏	Na	高速炉
40	development of NaK-to-mercury tube-in-tube, using chromium--molybdenum steel, 24:52975 (N-70-27200)	米	NaK-Hg	SNAP-8
41	development of power reactor, 24:29010	オーストリア		
42	development of reactor once-through steam generators, 24:46077	米		PWR FBR HTGR
43	dynamic behavior in power and desalination plants, analog model for simulation of, 24:38619	イスラエル		
44	dynamics for, digital computer calculation of, (T), 24:50236	ソ		
45	dynamics of two loop power reactor, effects of power transients on, (T), 24:33244	ソ		
46	efficiency of steam, for nuclear power plants, 24:39122	ソ		
47	evaporators for liquid sodium heated steam, heat transfer and pressure drop characteristics of, 24:52905	英	Na	
48	fabrication and operation of, recent developments in US, 24:50505	米		
26	fabrication of carbon dioxide cooled power reactor sectional, 24:28876	チェコ	CO_2	GCR
49	fabrication of prototype modular, 24:33294	米		FBR
50	failure behavior of high pressure, 24:29011			
51	flow diagram for power power reactor fossil fuel fired dual circuit superheat, 24:20596(P)			
52	flow stability in gas cooled reactor, 24:28985(R) (GA-8879)	米	He	Fort St. Vrain(HTGR)
53	fluid flow in natural circulation, hydrodynamic instability in, 24:36497	米		
54	heat exchanger for THTR, testing of helically wound, 24:36455 (JUL-649-RB)	西独		THTR
55	heat transfer in gas cooled reactor, 24:16227	西独		GCR

56	heat transfer in sodium heated, analog computer simulation of dynamic, 24:31091	西独		Na	SNR-300 (FBR)
57	heat transfer model for once-through sodium-heated, critical parameters for, 24:20946	伊		Na	
58	heat transfer system for liquid sodium cooled, operation of thermal convection tube, 24:36059	米		Na	
55	hydrodynamic resistance in gas cooled reactor, 24:16227	西独			GCR
59	inspection code for water cooled power reactor, ASME, 24:20481	米		水	LWR
60	inspection of prototype modular, 24:33294	米			
61	leak detection in LMFBR, development of hydrogen monitor for, 24:33405	米		Na	LMFBR
62	leak detection in sodium heated steam generators, evaluation of systems for, (E/T), 24:41018 (APDA-255)	米		Na	
63	leak detection systems for liquid sodium heated steam, 24:52904	米		Na	
64	leak detection systems for liquid sodium, development of, 24:52931	ソ		Na	
65	leaks in liquid sodium cooled reactor, detection systems for, (E), 24:43730 (APDA-257)	米		Na	
66	leaks in LMFBR, performance of acoustic emission systems for detection of, 24:38515 (APDA-256)	米		Na	LMFBR
67	leaks in sodium heated, detection systems for small, 24:9136 (APDA-TM-52)	米		Na	
68	level regulators for marine reactor, development of, (T), 24:9157 (RT/ING-(69)4)	伊			マリン
69	maintenance requirements and economics for liquid sodium cooled power reactors, 24:52900	英		Na	
70	materials for LMFBR, review of properties of, 24:32419	米	IHX	Na	LMFBR
71	materials for steam, corrosion of welds in, 24:44659 (ORNL-4560, pp 275-80)	米			
72	mounting of gas-cooled reactor, design of support for, 24:22854(P)				GCR
73	performance characteristics of once-through, 24:44161(T) (CONF-700211-4)	日			
74	performance of high temperature helium cooled reactor, 24:28875	英		高温He	Dragon(HTGR)
75	performance of NaK-to-mercury, with tantalum and stainless steel double-containment, (E), 24:52974 (N-70-26934)	米		NaK-水銀	SNAP-8
76	performance of once-through steam, with heated twisted tapes, 24:31577	伊			
77	performance testing of prototype LMFBR, 24:20936(R) (LMEC-70-1)	米		Na	LMFBR
78	pipes for, radiometric gage for detection of corrosion in, 24:50640	パラグアイ			
79	piping systems for, design and testing of sodium--water, 24:368	仏		Na-水	
80	repair in Marcoule power reactors, procedures for, 24:4007	仏			Marcoule
7	retubing radiochemical processing plant, equipment and methods for, 24:19124 (DUN-5935)	米			
81	seismic analysis of power reactor, (T), 24:13545	米			
82	simulation of sodium heated once-through, method for hybrid computer, 24:38536	オランダ			

83	stability of power reactor, use of hydraulic impedance for analysis of, (E/T), 24:7104	米		
84	stability of steam, effects of non-uniform heat flux on, 24:16771 (NP-18152)	米		
85	steam separators for, design of, 24:25189(P)			
86	stress in internally pressurized torospherical heads for, computer analysis of elastic, 24:31637	英		
87	stress in internally pressurized torospherical heads for, computer analysis of elastic--plastic, 24:31633	英		
88	support systems for pressurized water reactor, design of, 24:45846	米	水	PWR
38	testing liquid sodium heated steam, 24:52901	米	Na	LMFBR
89	testing of sodium-heated, for LMFBR design data, 24:14126(R) (LMEC-69-30)	米	Na	LMFBR
90	testing of sodium-heated, under simulated LMFBR conditions, 24:48796(R) (LMEC-70-13)	米	Na	LMFBR
91	testing program for liquid sodium heated steam, 24:52902	米	Na	
92	testing program for liquid sodium heated steam, 24:52903	日	Na	
93	thermodynamics for water--steam secondary system of gas heated, (E/T), 24:52913	仏	水	
94	tube failure in sodium-heated, safety hazards from pressurized steel, 24:10003 (TRG-Report-1932)	英	Na	
95	tube support structure for Fort St. Vrain Power Reactor Unit 1, analysis of sleeve--wedge test data for damping of, 24:11095 (GAMD-9735)	米	He	Fort St. Vrain(HTGR)
96	tube wastage in LMFBR, effects of water leaks on chromium--molybdenum--steel, 24:13521(R) (APDA-242)	米	Na	LMFBR
65	tube wastage in sodium--water, dominant system factors affecting, 24:43730 (APDA-257)	米	Na	
97	tube wastage in, effects of sodium--water reactions on, (E), 24:47853(R) (APDA-262)	米	Na	
98	tubes for liquid sodium, fabrication and inspection of stabilized steel, 24:52757	英	Na	PFR (FBR)
99	tubes for nuclear power plant, production and quality control of Inconel 600, 24:46709	日	水	CWR
100	tubes for pressurized water reactor, arrangement of helical strip spaced parallel, 24:24828(P)			PWR
101	tubes for sodium-cooled, review of properties of materials for, 24:32420	英	Na	LMFBR
102	tubes of liquid metal cooled reactors, effects of sodium and nitrogen on physical properties of ferritic and austenitic steel, 24:52895	西独	液体金属 Na	
103	weldments in, corrosion by steam of dissimilar metal, 24:17202 (ORNL-4480, pp 297-305)	米		
104	welds in high temperature gas cooled reactor steam generators, corrosion resistance of nickel alloy, 24:24835 (ORNL-4508, pp 3-70)	米	He	HTGR
105	welds in high temperature gas cooled reactor steam generators, corrosion resistance of nickel alloy, 24:4060 (ORNL-4424, pp 3-72)	米	He	HTGR

1971 Vol. 25

71000	BOILERS see also Superheaters			
1	components for, properties of welds in low-alloy, low-carbon steel, 25:49378 (ORNL-TM-3416, pp 143-7)	米		
2	construction aspects for marine reactors, 25:3918	ソ		マリンリアクター
3	construction of reactor, standards for, 25:43993 (RDT-E-15-2-T)	米		
4	corrosion of nuclear power plant steel, methods for minimizing, 25:52294 (A/CONF.49/P-576)	仏		
5	description of integrated power reactor, 25:34396(P)			
6	design and operation of sodium-heated, in US, review of, 25:39621 (CONF-710315-10)	米	Na	
7	design characteristics of power reactor, 25:3898	西独		GCR,LWR, FBR
8	design characteristics of superheating reactor core vessel, 25:60147(P)	—		
9	design codes and standards in U.S. for liquid metal cooled fast breeder reactor, 25:20723 (SAN-781-217)	米		LMFBR
10	design criteria for pressurized water reactor, analysis of aseismic. (E/T), 25:10248	米	水	PWR
11	design for indirect-cycle nuclear power plants, survey of, 25:8113	スイス		
12	design for LMFBR, modular, 25:44483 (CONF-710316-26)	米		LMFBR
13	design for molten salt and liquid metal cooled reactor, 25:28746(P)			NSRE,LMFBR
14	design for radioisotope-fueled organic Rankine cycle power system, 25:18594(R) (ORNL-TM-3292)	米		ランキンサイクル
15	design of compact, for pressurized water reactors, 25:60144(P)			PWR
16	design of forced flow steam, heated by pressurized water from nuclear reactors, 25:54836(P)			PWR
17	design of liquid metal-heated, review of feasibility, 25:2600 (BNL-50258)	米	液体金属	
18	design of nuclear power plant, calculation methods for, 25:15815	米		
19	design of steam generator, for enclosure in pressure vessels, 25:46287 (EUR-4336)	西独	He	THTR
20	design of steam generators for molten salt reactors, thermal stress and salt freezing minimization in, (T), 25:39282 (ORNL-TM-2953)	米		MSRE
21	design of steam, for reactor power plants, calculation methods for, 25:308	西独		
22	design parameters for 330 MW(e) high temperature gas cooled reactor, 25:28631	米	He	Fort St. Vrain(HTGR)
23	design parameters for high temperature gas cooled power reactors, computer calculations for optimum, 25:1916	西独	He	HTGR
24	design parameters of bayonet-tube split bundle liquid sodium heated, for 1000 MW(e) liquid sodium cooled fast breeder reactor, 25:51155	米	Na	LMFBR
25	design parameters of gas cooled power reactor, 25:25956	米	He	Fort St. Vrain(HTGR)
26	design parameters of liquid sodium heated, for demonstration 350 MW(e) liquid metal fast breeder reactor, 25:41498 (CONF-710315-4)	米	Na	LMFBR

27	design parameters of liquid sodium heated, for liquid metal cooled fast breeder reactors, 25:41499 (CONF-710315-16)	米	Na	LMFBR
28	design parameters of power reactor, 25:28748	西独		
29	design parameters of pressure vessel enclosed, for pressurized water reactors, 25:28751(P)			PWR
30	design parameters of prototype 26.7 MW(t) liquid sodium heated, 25:39281 (CONF-710315-2)	米	Na	LMFBR
31	design parameters of sodium heated steam generators, with split bundles of bayonet tubes, 25:46286 (CONF-710315-24)	米	Na	LMFBR
32	design problems for prototype liquid sodium cooled fast breeder reactors, 25:18115	日	Na	LMFBR
33	design problems with fast reactor liquid sodium heated, 25:60077	仏	Na	Phenix (LMFBR)
30	design requirements for 1000 MW(e) liquid sodium heated, 25:39281 (CONF-710315-2)	米	Na	LMFBR
34	development for sodium cooled fast breeder reactors, review of, 25:23324	米	Na	LMFBR
35	development in Japan of liquid metal heated, for fast breeder reactors, 25:18133	日	Na	LMFBR
36	development in U.S. of high temperature gas cooled power reactor, review of, 25:31623 (GA-10338)	米	He	HTGR
37	development of gas cooled power reactor, 25:3899	英		GCR
38	development of gas cooled reactor, 25:10166 (GA-10265)	米		GCR
39	development of gas cooled reactor, 25:18099(R) (GA-10313)	米		GCR
40	development of module type, for liquid sodium cooled fast breeder reactors, 25:23326	日	Na	LMFBR
41	development of once-through steam generator, review of test results on, 25:31532 (BAW-10027)	米		
42	development of sodium heated modular steam generator concept, 25:23323 (TID-25669)	米	Na	
30	development problems for 1000 MW(e) liquid sodium heated, 25:39281 (CONF-710315-2)	米	Na	
25	fabrication of gas cooled power reactor, 25:26956	米	He	HTGR
43	fabrication of gas cooled reactor, 25:12766	仏	CO_2	EL-4(GCR)
44	failure of sodium-heated, effects of sodium corrosion on, 25:49357 (ORNL-TM-3300, pp 182-7)	米	Na	
45	feedwater storage for nuclear power plant, development of heated rock cavities for, 25:53917 (A/CONF.49/p-798)	スウェーデン		
46	flow in curved tubes in, heat transfer in two-phase, 25:18586	日		
17	flow patterns in mercury, review of x-ray evaluations of, 25:2600 (BNL-50258)	米		
47	heat transfer characteristics of once-through water heated, effects of design parameters on, 25:18252	西独		
48	heat transfer coefficients for once-through high pressure, effects of steam quality on, 25:13149 (EUR-4561)	伊		
49	heat transfer in forced-convection water, effects of surface crud deposits on, (E), 25:54861 (AEEW-R-705)	英		
50	heat transfer in sodium-heated, relations of critical and Laidenfrost point, 25:4211	米	Na	

51	inspection and testing of failed sodium-heated, status of, 25:49360 (ORNL-TM-3416, pp 155-65)	米	Na	
52	inspection of welded T-joints in, x-radiographic method for, 25:34840	日		
53	instrumentation for pressurized water reactor, review of process, 25:37010 (WCAP-7671)	米	水	PWR
54	joints in, thermal shock testing of tantalum--stainless steel, 25:54225 (N-71-22511)	米		SNAP-8
55	leak detection in, 25:39470(R) (ANL-7750)	米		
56	leak detection in sodium heated, design of hydrogen meter for, 25:60058(R) (ANL-7817)	米	Na	FFTF
57	leak detection in sodium heated, hydrogen monitor for, (E/T), 25:51111	米	Na	LMFBR
58	leak detection system for sodium-heated, design of, 25:49350 (APDA-265)	米	Na	
59	leak detectors for liquid sodium heated, development of hydrogen monitoring, (E/T), 25:1878(T) (ANL-Trans-857)	ソ	Na	
60	leak detectors for steam, development of ion pump-type, 25:4736(R) (ANL/ST-3)	米		
61	leaks in sodium-heated, development of detection methods for, 25:13651(R) (ANL/ST-4)	米	Na	
62	leaks in sodium-heated, development of detection methods for, 25:37918(R) (ANL/ST-8)	米	Na	
63	leaks in sodium-heated, evaluation of systems for detecting, 25:57632	英	Na	
64	leaks in welded nuclear power plant, comparison of ammonia and helium test methods for detecting, 25:6502	仏		
65	magnetite coatings in, relations of magnetite morphology with pressure drop, 25:58165(T) (AEC-tr-7237)	西独		
66	materials for steam development of advanced, 25:8838 (ORNL-4600, pp 282-6)	米		
67	materials for, corrosion testing plans for dissimilar metal-couple welds in, 25:29763 (ORNL-4630, pp 280-7)	米		
27	operation aspects of liquid sodium heated, for liquid metal cooled fast breeder reactors, 25:41499 (CONF-710315-16)	米	Na	LMFBR
68	operation of sodium-heated prototype, under simulated LMFBR conditions, 25:42804(R) (LMEC-71-1)	米	Na	LMFBR
69	packaging and shipping of radioactive, in Canada, development of special arrangement methods for, 25:34843 (AECL-3804)	カ		
70	performance of Alco/BLH prototype once-through sodium-heated, evaluation of structural and thermohydraulic, 25:51673 (LMEC-Memo-70-20)	米	Na	
71	performance of once-through NaK heated mercury, 25:60194 (N-71-27855)	米	水銀-NaK	SNAP-8
72	performance of short reentry tube steam for use in radioisotope power plant, effects of configuration, flow rate, and pressure on, 25:41962 (ORNL-TM-3236)	米		
73	performance of sodium-heated, in large component test facility, evaluation of, 25:42804(R) (LMEC-71-1)	米	Na	
74	pressure distribution in helium--water, MEDEA code for time independent, 25:31624 (JUL-698-RG)	西独	He	
75	pressure transients in liquid sodium heated, calculation of sodium-water reaction, 25:41500	米	Na	

76	reliability analysis of power reactor, 25:5982 (RISO-214, pp 301-22)	西独		
77	safety analysis of liquid sodium heated, for liquid metal cooled reactors, (E/T), 25:39280 (CEA-N-1392)	仏	Na	LMFBR
78	standards for steam generators for pressurized water power reactors, 25:44081 (RDT-E-4-1-T)	米		PWR
79	steam quality in power, rhenium-186 tracer method for measuring, 25:57674	ポーランド		
80	stress analysis of power reactor, finite element method for, 25:60142	日		
81	testing for thermoelectric converters, 25:10778(R) (ORNL-TM-3214)	米		
25	testing of gas cooled power reactor, 25:26956	米	He	Fort St. Vrain(HTGR)
43	testing of gas cooled reactor, 25:12766	仏	CO_2	EL-4(GCR)
84	testing of gas cooled reactor, pressure and temperature, 25:36978(R) (GA-10560)	米		GCR
85	testing of sodium-heated, description of facility for, 25:13143	仏	Na	
86	tube wall temperatures in once-through sodium-heated, effects of heat flux, mass velocity, pressure, and tube diameter on, 25:51670 (AEEW-M-1000)	英	Na	
87	tubes for Dragon reactor, properties of materials for, 25:4748 (DP-Report-537)	英	He	Dragon(HTGR)
88	tubes for LMFBR, properties of candidate alloys for, 25:39283	米		LMFBR
89	tubes for sodium heated steam generators, properties of carbon steels for, 25:34334(T) (JPRS-53069, pp 188-96)	ソ	Na	
90	tubes for steam hydrodynamic stability in curved, 25:4208	米		
91	tubes for, evaluation of Hastelloy N, 25:16399 (ORNL-4622, pp 119-98)	米		
92	welds in high temperature gas cooled reactor, corrosion resistance of nickel alloy, 25:5990(R) (ORNL-4589)	米	He	HTGR

1972 Vol. 26

72000	BOILERS see also Superheaters			
1	air flow characteristics for gas cooled reactor, 26:6232(R) (GULF-GA-A-10754)	米	air	GCR
2	air flow test for helium heated, full scale, (E/T), 26:52551 (GULF-GA-A-10833)	米 米	air-He	
3	blowdown of saturated steam--water-filled, with multiple release apertures, safety testing of, 26:47983 (CISE-R-303)	伊		
4	book: Materials of Construction for Nuclear Reactors. Part I. Nuclear and Thermophysical Properties, Elements of Corrosion and Heat Stability (In Russian), 26:33890	ソ		
5	capacity of U.S.S.R. power reactor, methods for increaseing unit, 26:42506	ソ		
6	characteristics of helical, for gas-cooled reactors, 26:58319	ス	CO_2,He	THTR,HTGR, GCR
7	components of gas cooled reactor, performance under operating conditions of, 26:49843(R) (GULF-GA-A-12030)	米		GCR
8	construction of pressurized water reactor steam generator, standards for, 26:14483 (RDT-E-4-1-T(12-71))	米	水	PWR
9	corrosion by two-phase flowing water, 26:33893 (INIS-mf-213)	仏		
10	corrosion by water, mechanisms of iron and steel, 26:20791	英		
11	corrosion in, effects of water-surface deposits in tubes on, (E), 26:52690 (AE-438)	スウェーデン		
12	corrosion of carbon steel, by feed water, hydrogen and oxygen content control for inhibition of, 26:41370(P)			
13	corrosion of steel, by moist carbon dioxide at 350 to 430°C, 26:20790	西独	CO_2	
14	corrosion of weldments in, preparation of specimens for testing, 26:15673(R) (ORNL-TM-3550)	米		
15	deposits chemistry in HTGR, 26:45288 (TID-26106)	米	He	HTGR
16	description of once-through, for increased cycle efficiency of pressurized water reactors, 26:58320	米	水	PWR
17	description of pressure vessel enclosed, for pressurized water reactors, 26:3825(P)		水	PWR
18	design and operation of once-through steam technology course on, 26:9073 (GAMD-10457)	米		
19	design characteristics of 300 MW(e) prototype steam cooled fast breeder reactor, 26:3935 (BMBW-FBK-71-4)	西独		FBR
20	design characteristics of marine reactor, analog simulation of, 26:22352	仏		マリン
21	design characteristics of prestressed concrete pressure vessel contained, for gas cooled reactors, 26:16890	西独		GCR
22	design characteristics of vertical shell and tube, for liquid metal cooled reactors, 26:35462(P)		液体金属	
23	design for liquid metal-cooled fast reactors, 26:58321	英	Na	LMFBR
24	design for radioisotope-fueled electric power generator, 26:4472(P)			
25	design for sodium-cooled reactors, 26:238	英	Na	

26	design for use with radioisotope-fueled heat source, 26:4473(P)	ソ		
27	design of 1 MW liquid sodium--water steam generator, for leak tests, 26:55153 (CONF-710548-. pp 57-104)	ソ	Na	
28	design of cylindrical liquid metal-heated, subjected to transient internal loads, review on methods for, 26:22760	米	液体金属	
29	design of divided steam generators, for high temperature gas cooled reactors, 26:6320	英	He	HTGR
30	design of liquid metal-heated, 26:22715(P)		液体金属	
31	design of once-through steam, for nuclear power plant, 26:33272 (INIS-mf-200)	米		
32	design of power reactor, with interconnected economizer, evaporator, and superheater tube sections, 26:35423(P)			
33	design of sodium--water, for LMFBR, 26:52366 (INIS-mf-234)	伊	Na	LMFBR
34	design of sodium-heated steam, 26:45450(P)		Na	
35	design parameters of molten salt breeder reactor, 26:24754 (ORNL-4728, pp 1-44)	米		MSBR
36	design with water/steam reheat features, for gas cooled reactors, 26:11386(P)			GCR
37	development for fast breeder reactors, design of sodium loop facility for, 26:2036	仏	Na	FBR
38	development of integral economizer once through steam generator, 26:58086 (BAW-10027(Suppl.1))	米		
39	development of modular sodium heated steam generators, 26:24779	米	Na	LMFBR
40	development of oil-fired potassium, 26:52415	米	K	
41	development of sodium--water steam generator, test programs for, 26:55155 (CONF-710548-, pp 129-54)	日	Na	LMFBR
42	development of U-shaped tube bundle, for liquid sodium cooled fast power reactors, 26:35463	チェコ	Na	LMFBR
43	development program for molten salt breeder reactor, 26:35434 (CONF-710443)	米		MSBR
44	dynamics for liquid sodium heated once-through steam generators, simulation of, 26:37850	オランダ	Na	
45	fabrication of reactor, methods for, 26:20823	日		PWR
46	fabrication of water reactor, selection of steel for, 26:23366	米	水	LWR
47	feedwater for pressurized water reactors, standards for, 26:42505	ソ	水	PWR
48	flow in once-through, mathematical model for describing transients in two-phase, 26:47981 (CEA-CONF-2009)	仏		
49	inspection of Alco/BLH, post-failure, 26:2042	米		
50	inspection of failed ALCO/BLH sodium heated, 26:4912 (ORNL-4770, pp 97-9)	米	Na	
14	inspection of failed ALCO/BLH, for analysis of corrosion mechanisms in sodium heating tests, 26:15673(R) (ORNL-TM-3550)	米	Na	
51	inspection of failed Alco/BLH, p. 150-156, 26:36489(R) (ORNL-TM-3703)	米		
52	interactions with steam turbines, model for simulation of dynamic, 26:60737 (FRNC-CONF-50)	仏		
53	leak detection in sodium heated, development of systems for, 26:32764(R) (ANL-7905)	米	Na	

54	leak detection in sodium heated, hydrogen meter for, 26:3877	米	Na	
55	leak detection in sodium heated, hydrogen meter for, 26:3878	米	Na	
56	leak testing of straight tube, parameters for large, 26:55151 (CONF-710548-, pp 17-38)	西独	Na	SNR
57	leakage of water into sodium in steam generator, damage and effects on safety of, 26:55150 (CONF-710548-, pp 1-16)	仏	Na	LMFBR
58	leaks in LMFBR, detection of, p. 54-55, 26:28609 (ANL-7775, pp 37-63)	米	Na	LMFBR
59	leaks in sodium--water steam generators, use of hydrogen detector for detection of small, 26:55156 (CONF-710548-, pp 155-70)	日	Na	LMFBR
60	leaks in, development of detectors for, 26:30666(R) (ANL-7868)	米		
27	leaks of water into liquid sodium in 1 MW steam generator, corrosive effects on tube bundles of, 26:55153 (CONF-710548-, pp 57-104)	ソ	Na	
61	leaks of water into sodium in steam generator, effects of reactions from, 26:55154 (CONF-710548, pp 105-28)	ソ	Na	
62	operating characteristics of pressurized water reactor, 26:1598	米		PWR
63	operation and performance sodium-heated, 26:27999	日	Na	
64	operation of dual purpose production and power reactor, analog computer simulation of, 26:35622 (HW-77921-F)	米		
65	operation of mercury, temperature transients during startup and shutdown, 26:52599 (NAA-SR-Memo-9726)	米		
27	performance of 1 MW sodium--water steam generator, 26:55153 (CONF-710548-, pp 57-104)	ソ	Na	
66	performance of exhaust heat recovery, operating in Rankine cycle, computer program for simulating, 26:55950 (K-1837)	米		ランキンサイクル
67	performance of single tantalum tube mercury, 26:30377 (N-71-35796)	米		SNAP
68	performance of sodium--water steam generator, 26:55159 (CONF-710548-, pp 206-12)	ソ	Na-水	
69	pressure barriers for gas heated, 26:37847(P)			
70	pressure drop in forced-flow alkali metal and water effects of helical-flow inserts on, (E), 26:25489	米		
71	pressure in sodium heated, effects of sodium--water reactions on, (E/T), 26:42515	西独	Na	
72	pressure transients in sodium heated steam generators, effects of sodium-water reactions on, (T), 26:32789	米	Na-水	
73	production of standarad, for Westinghouse PWRs, 26:58318	米	水	PWR
74	pumps for nuclear steam, design of, 26:6744(P)			
75	safety aspects of sodium--water reactions in fast breeder reactor, 26:14509	日	Na	FBR
68	safety of sodium--water steam generators, 26:55159 (CONF-710548-, pp 206-12)	ソ	Na	
76	shipping to reactor sites, equipment and methods for, 26:2034	米		
77	standards for sodium heated, 26:49946 (RDT-E-4-16-T(5-72))	米	Na	
78	startup of mercury, feasibility of repeated cyclic, 26:52598 (NAA-SR-Memo-9725)	米		

79	steam generators for power reactors, emergency feedwater supply system for, 26:14496(P)			
80	steam pressure control in, design and testing of system for, 26:40260	カ		
81	steam pressures in dual purpose production and power reactors, effects of plugged tube combinations on secondary, 26:35623 (HW-78194)	米		
82	stresses in LOFT steam, review on initial analyses of, 26:22711 (TID-25925)	米		
83	survey of nuclear, 26:58317			
84	testing of once-through, for FFTF and LMFBR, p.1-1, 26:31450(R) (LMEC-72-1)	米	Na	FFTF,LMFBR
85	thermal characteristics of, for reactor cooling system, MAIN computer program for calculating, 26:25456 (KFK-715)	西独		
86	thermodynamic analysis of, computer program for, 26:62365(T) (EURFNR-983)	西独		
87	tube failure in austenic steel steam generator, effects of sodium--water reactions on, 26:55161 (CONF-710548-, pp 221-42)	ソ		
88	tube lifetime of high temperature gas cooled power reactor, effect of oscillating dryout point on, 26:44762	米	He	HTGR
89	tube sheet welds in, radiographic inspection of, 26:9181	日		
90	tube wall temperatures in evaporation region of subcritical onec-through sodium-heated, method for estimating, 26:9103 (AEEW-M-1000)	英	Na	
91	tube wastage in sodium-heated, method for evaluation of Croloy, 26:56643(T) (JAPFNR-25)	日	Na	
92	tubes for gas cooled, sealing means for concentric, 26:16869(P)			
93	tubes for Public Service of Colorado reactor power plant, air flow-induced vibration testing of, 26:2136	米	He	HTGR
94	tubes for, development of ferritic steel, 26:9735(R) (UCLA-34-P-177-7)	米		
95	tubes for, internal TIG welding method for fabrication of, 26:18243	日	Na	LMFBR
96	tubes for, pressure loss in helical steel, (E), 26:8559	西独		GCR
97	tubes in, stress corrosion cracking of stainless steel, (E), 26:1635 (DUN-SA-178)	米		Hanford
98	tubes of gas cooled reactor, carbon deposition on, (E), 26:42519	西独	CO_2	HTR, GCR
99	water chemistry in, review on, 26:33123 (WARD-5788)	米		
100	water level in ship, subjected to sea-wave motion, system for regulating, 26:17661 (EUR-4692)	伊		
101	welded joints in large-capacity, radiometric method for inspection of, 26:45473	ソ		
102	weldments on, corrosion by chloride-containing steam, p. 142-50, 26:36489(R) (ORNL-TM-3703)	米		

1973 Vol. 27

73000	BOILERS see also Superheaters			
1	blower for increased plant kilowatts, design of turbine-driven, 27:29814	米		
2	book: Steam Generator Installations for Nuclear Electric Stations, 27:19134	ソ		
3	cleaning of, uses of complexones in, 27:11744(T) (CONF-701074-12)	ソ		
4	conference on corrosion of steam, Ermenonville, France, March 13, 1972, 27:5617	仏		
5	control system for once-through steam generators for high temperature gas-cooled reactors, 27:23875(T) (LIB/Trans-497)	伊		HTGR
6	corrosion prevention in U-tube, for pressurized water reactors, 27:26835	米	水	PWR
7	corrosion testing of steam, for pressurized water reactors, 27:5618	仏	水	PWR
8	description of, for nuclear power plants, 27:5016			
9	design and materials for HTGR-type reactors, 27:14040	米	He	HTGR
2	design characteristics of power reactor, 27:19134	ソ		
10	design characteristics of pressurized water marine propulsion reactor, 27:16773 (CONF-710532-(Vol.2), pp 563-580)	伊	水	マリン PWR
11	design characteristics of U-tube steam one-dimensional model for studying, 27:24523 (RISO-M-1564)			
12	design for feedwater vaporization in superheating reactor, 27:11395(P)			
13	design of 500 MW(t) single-pass, 27:16512(T) (CONF-701074-9)	ソ		
14	design of high-integrity sodium-heated, 27:16595	英	Na	
15	design of high-pressure modular heat recovery, for gas turbine--steam turbine combined cycles, 27:29813	米		
16	development for 1000 MW(e) power plants with water-moderated water-cooled reactors, 27:14070(T) (CONF-701074-3)	ソ	水	LWR
17	development in Italy of liquid sodium heated, 27:16565	伊	Na	
18	development in U.K. of nuclear power plant, survey of, 27:11397	英		
19	development of integrated heat exchanger--, for sodium cooled fast breeder reactors, 27:23904 (BMBW-FBK-72-28)	西独	Na	LMFBR
20	development of steam generators for AGR-type reactors, problems in, 27:4420	英	CO_2	AGR
21	dynamic analysis of once-through, comparison of PANDYNE computer code and test results of, 27:22411	スイス		
22	experience with gas heated, for nuclear power plants, 27:11475	スイス	Gas	
23	feedwater for, effects of structural materials on optimum temperature of, 27:16741(T) (JPRS-56318, pp 10-20)	ソ		
24	feedwater pumps for AGR-type reactor, design of, 27:21647	英	CO_2	AGR
25	feedwater pumps for power reactor, development of, 27:21646	英		
26	feedwater pumps for power reactor, development of single-stage, 27:21649	英		

27	feedwater pumps for power reactor, reliability criteria for, 27:21648	英		
28	feedwater pumps for water cooled reactor, development of single stage, 27:21650	米	水	LWR
29	flow and heat transfer characteristics of large once-through steam for MSBR, dynamic model for studying, 27:17179	米		MSBR
30	fluid flow in steam, effects of tube configuration on two-phase, 27:5189	仏		
31	gaseous wastes from industrial, 27:24579	米		
32	heat pipes for sodium cooled reactor steam generators, performance of, (T), 27:26831 (EURFNR-1069)	西独	Na	LMFBR
33	heat transfer characteristics of sodium cooled reactor, 27:29523	仏	Na	LMFBR
34	inspection of failed ALCO/BLH sodium-heated, 27:725(R) (ORNL-TM-3969)	米	Na	
35	materials for nuclear power plant, effects of thickness and pressure on plastic deformation of, 27:5676	米		
36	materials for SNAP-8 mercury, evaluation of stainless steel--tantalum as containment, (E), 27:11485	米		SNAP-8
37	materials for SNAP-8 mercury, long-term mock-up testing of, 27:9948	米		SNAP-8
38	materials for SNAP-8, fabrication and testing of tantalum as mercury containment, 27:9940	米		SNAP-8
39	materials for steam properties of corrosion-resistant steel, 27:5620	仏		
40	materials for, high-temperature compatibility of, 27:5649 (ORNL-4820, pp 38-95)	米		
41	nitrogen oxide release from, methods for reduction of, 27:24578	米		
42	operation and inspection of sodium-heated steam generator test model, 27:26753 (GEAP-10580-1)	米	Na	
2	performance of power reactor, 27:19134	ソ		
43	performance of prototype 45 MW sodium heated steam generator, thermal and hydraulic, 27:2008 (FRNC-CONF-42)	仏	Na	
44	performance of sodium-heated steam generator test model, thermal-hydraulic, 27:26754 (GEAP-10580-2)	米	Na	
45	performance of steam generator, digital model for dynamic, 27:2007 (FRNC-CONF-37)	仏		
46	pipe failures in sodium heated, effects on reactor cooling system of sodium-water reactions during, (E/T), 27:19086(T) (CONF-701074-6)	ソ	Na	LMFBR
47	porous deposits on heat transfer surfaces of fossil- and nuclear-fueled power plant, heat and mass transfer in, 27:11924 (WARD-5836)	米		
48	radiation monitoring in primary circuit of water-cooled reactors, 27:19090	ソ	水	LWR
49	review of pressurized water reactor type, 27:16743	西独	水	PWR
43	safety of prototype 45 MW sodium heated steam generator, simulation of effects of sodium--water reactions in, 27:2008 (FRNC-CONF-42)	仏	Na	FBR
50	stress analysis of, development of TRANS computer program for dynamic, 27:2152	オーストリア		
51	support structures for pressurized water reactor, design of, 27:29476(P)		水	PWR
52	testing of in-line tube and ribbed helical, for gas-cooled reactors, 27:14041	英	He-air	Dragon (HTGR)
53	tubes for, effects of surface deposits on heat transfer properties of, 27:17183 (AE-459)	チェコ		
54	tubes for, inspection of Inconel 600, 27:14285 (DUN-7794)	米		
55	water chemistry in, 27:5017			
3	water quality in, use of complexones for improving, 27:11744(T) (CONF-701074-12)	ソ		

Steam generators 1973 Vol. 28

73000	STEAM GENERATORS			
56	Steam generator and heat exchanger materials (LMFBR), 28:8624 (ORNL-TM-4105)	米		LMFBR
57	Annual review of steam generator technique. Second edition, 1972/73 (Nuclear power plants), 28:1561	西独	水	PWR
58	STEAM GENERATORS/BLOWDOWN Moisture carryover during an NSSS steam line break accident, 28:26598 (CENPD-80(Rev.1))	米		
59	STEAM GENERATORS/CORROSION PROTECTION Preservation of steam boilers, 28:17709 (AEC-tr-7435)	西独		
60	STEAM GENERATORS/DESIGN Applying the experience gained in nuclear power plant development to non-nuclear power and process plants, 28:10195	英		
61	Development of steam generator for liquid metal fast breeder reactor, 28:17452	日	Na液体金属	LMFBR
62	Installation for vapor production for a nuclear reactor, 28:17583 (P)			
63	Main components in high power fast breeder reactors, 28:31829	仏	Na	Phenix(FBR)
64	Once-through steam generation, 28:9980	英	水	PWR
65	Reliability of fast reactor steam generators, 28:31833	英	Na	高速炉
66	Steam generators for sodium-cooled fast breeder reactors. State of development and design guide, 28:1619	日	Na	FBR
67	Supporting device for tubular arrays helically wound to form several coaxial cylindrical layers, 28:17581 (P)			
68	STEAM GENERATORS/EXPLOSION WELDING Current Canadian use of explosive welding for repair and manufacture of nuclear steam generators, 28:12064 (AECL-4427)	カ		
69	STEAM GENERATORS/FABRICATION Applying the experience gained in nuclear power plant development to non-nuclear power and process plants, 28:10195	英		
70	Development of steam generator for liquid metal fast breeder reactor, 28:17452	日	液体金属	LMFBR
71	STEAM GENERATORS/FAILURES Steam generator tube failures: a world survey of water cooled nuclear power reactors to the end of 1971, 28:20370 (AECL-4449)	カ	水	LWR
72	STEAM GENERATORS/FEEDWATER Experiments on chemical degassing of feed water for steam generators (Using hydrazine), 28:32135 (AEC-tr-7447)	西独		
73	STEAM GENERATORS/FLUIDIZED BED Fluidized bed coal combustion system coupled to a potassium vapor cycle (Engineering Materials), 28:26797 (CAPE-2351)	米		
74	STEAM GENERATORS/HEAT TRANSFER Application of nuclear two-phase thermohydraulic developments to non-nuclear evaporator systems, 28:8098	スイス		
75	STEAM GENERATORS/LEAKS Leakage of water into sodium in steam generators and test rigs (LMFBR), 28:10015	米	Na	LMFBR
76	STEAM GENERATORS/NONDESTRUCTIVE TESTING Nondestructive testing for steam generators, 28:12819	米		

64	STEAM GENERATORS/PERFORMANCE Once-through steam generation, 28:9980	米		
77	Steam: its generation and use (Book), 28:15436			
70	STEAM GENERATORS/PERFORMANCE TESTING Development of steam generator for liquid metal fast breeder reactor, 28:17452	日	液体金属	LMFBR
78	Test requirements and testing of pressure systems for nuclear power stations. Experience with the THTR liner and steam generators, 28:20423	西独	He	THTR
79	STEAM GENERATORS/RELIABILITY Reliability of fast reactor steam generators, 28:31833	英	Na	PFR(FBR)
69	STEAM GENERATORS/RESEARCH PROGRAMS Applying the experience gained in nuclear power plant development to non-nuclear power and process plants, 28:10195	英		
80	STEAM GENERATORS/SEALS Monitoring the seal of the primary loop in steam generators at nuclear power plants with a water-moderated water-cooled reactor, 28:23410 (JPRS-58858)	ソ	水	LWR
81	STEAM GENERATORS/STRESS ANALYSIS Experimental and theoretical stress analysis in a nuclear steam generator head, 28:14412 (EUR-4820(Vol.3)(Pt.F))	伊		
82	STEAM GENERATORS/STRESS CORROSION Steam generator and heat exchanger materials, 28:11218 (ORNL-TM-4055)	米	Na	LMFBR
83	STEAM GENERATORS/SUPPORTS Dynamic analysis of nuclear equipment supports, 28:14591 (EUR-4820(Vol.3)(Pt.F))	米		
84	STEAM GENERATORS/TRSTING Liquid Metal Engineering Center (LMEC) annual technical progress report, January--December 1972 (LMFBR sodium component and instrumentation testing), 28:14422 (LMEC-73-1)	米	Na	LMFBR
85	STEAM GENERATORS/TUBES Dimensioning of components according to statistical criteria, 28:10196	オーストリア		
86	STEAM GENERATORS/TWO-PHASE FLOW Review of two-phase flow instability, 28:26478	仏		
87	STEAM GENERATORS/WELDED JOINTS High integrity tube-to-tubeplate bore welds, 28:23499			HGR

1974 Vol. 29

74000	STEAM GENERATORS Steam generator for pressurized water nuclear reactors (Patent), 29:17561		水	PWR
1	Steam generators for nuclear power plants, 29:12162	西独		
2	STEAM GENERATORS/COMBINED CYCLES Steam generator designs, 29:23811	米		
3	STEAM GENERATORS/CONFIGURATION New steam production concept for gas-cooled reactors, 29:26056	西独		GCR
4	Vertical or horizontal steam generators in nuclear power plants with pressurized water reactors, 29:14603	西独		PWR
5	STEAM GENERATORS/CONTROL EQUIPMENT Power from coal. III. Combustion controls and pollution controls, 29:31497			
6	STEAM GENERATORS/CORROSION Proceedings of the symposium on corrosion control by design in CANDU nuclear power reactors held at Power Projects on October 18, 1972, 29:17551 (AECL-4429)	カ	重水	CANDU (PHWR)
7	STEAM GENERATORS/CYCLONE SEPARATORS Apparatus for drying steam in nuclear power steam generators plants (Patent), 29:12152			
8	STEAM GENERATORS/DECONTAMINATION Iodine decontamination factors during PWR steam generation and steam venting, 29:4520 (CENPD-67)		水	PWR
9	STEAM GENERATORS/DEPARTURE NUCLEATE BOILING Axial heat flux effects on DNB in forced convection, 29:5032	米		
10	Temperature regime of a steam-generating surface on departure from nucleate boiling (DNB), 29:15496	ソ		
11	STEAM GENERATORS/DESIGN Experimental studies on flow induced vibration to suppo steam generator design. Part II. Tube vibration induced by liquid cross-flow in the entrance region o a steam generator, 29:2592 (AECL-4515)	カ		
12	Experimental studies on flow induced vibration to suppo steam generator design. Part I. Vibration of a heated cylinder in two-phase axial flow, 29:2591 (AEC 4514)	カ		
13	Large HTGR steam generator design verification and support programs. Summary, 29:28639 (GULF-GA-A-12808(Vol.1))	米	He	HTGR
14	Problems of design management in power plant erection: illustrated by the concept of steam generatots in pressurized water reactor nuclear power plants, 29:146		水	PWR
15	Reaction-protected steam generator for sodium-cooled reactor installations, 29:26255	西独	Na	
16	Steam generator designs, 29:23811	米		
17	STEAM GENERATORS/FABRICATION Fuels and Materials Development Program quarterly progress report for period ending March 31, 1973, 29:9 (ORNL-TM-4250)	米		
18	STEAM GENERATORS/FAILURES Method of characteristics for pressure transients in cylindrical geometry with axial symmetry: the formulation (LMFBR), 29:17571 (ANL-CT-73-23)	米		LMFBR
19	Review contribution on the phosphate regime in nuclear steam generators, 29:17560	西独		

20	STEAM GENERATORS/FEEDWATER Improvements in and relating to once-through boilers (Patent), 29:23446			
21	STEAM GENERATORS/FLUID FLOW Nuclear reactor boiler system (Patent), 29:17725			
22	STEAM GENERATORS/HEAT TRANSFER Area of departure from nucleate boiling in once-through steam generator, 29:5024	チェコ		
23	Mixed convection in the shell of a horizontal Na-to-NaK heat exchanger, 29:313		Na-NaK	
24	Out of pile facility for heat transfer tests of nuclear plant components prototypes, 29:18257 (RT/ING-(73)13)	伊		
25	Post dryout heat transfer in nuclear and cryogenic equipment, 29:2597 (AERE-R-7519)	英		
26	STEAM GENERATORS/HYDRAULICS Gas-cooled reactor programs annual progress report for period ending December 31, 1972, 29:23442 (ORNL-4911)	米		GCR
27	STEAM GENERATORS/LEAK DETECTORS Detection of leaks in the primary circuit of steam generators at nuclear power stations with water cooled and moderated reactors, 29:7251	ソ	水	LWR
28	Monitoring of the sodium--water leak-tightness of the sodium-heated steam generators by hydrogen dosage industrial application to the C.G.V.S., 29:23493 (AEC-tr-7493)	仏	Na	
29	Monitoring the sodium-water-tightness in sodium-heated steam generators by measuring hydrogen. Industrial application at CGVS, 29:7112	仏	Na	
30	STEAM GENERATORS/LEAK TESTING Evaluation of steam-to-sodium leak detectors on the SCTI, Phase II, 29:12658 (ANL-CT-73-27)	米	Na	LMFBR
31	STEAM GENERATORS/LEAKS Hydrogen-meter leak detector for LMFBR steam generators, 29:11857 (ANL-8047)	米		LMFBR
32	STEAM GENERATORS/MECHANICAL VIBRATIONS Experimental studies on flow induced vibration to support steam generator design. Part I. Vibration of a heated cylinder in two-phase axial flow, 29:29239 (CCNP 730465-)	米	水	
33	Experimental studies on flow induced vibration to support steam generator design. Part II. Tube vibration induced by liquid cross-flow in the entrance region of a steam generator, 29:31198 (CONF-730465-)	カ		
34	Large HTGR steam generator design verification and support programs. Vibrations, 29:28641 (GULF-GA-A-12808(Vol.4))	米	He	HTGR
35	PSC steam generator hot flow test (HTGR), 29:14611 (GULF-GA-A-12764)	米	He	HTGR
36	Vibration analysis in the design of steam generating equipment for the Bruce nuclear power station, 29:31199 (CONF-730465-)	カ	重水	Bruce (HWR)
37	Vibration of tube banks by wake force, 29:31177 (CONF-730465-)	日	CO_2	TOKAI(GCR)
38	STEAM GENERATORS/MEETINGS Conference on recent developments in compact high duty heat exchangers, London, England, October 12, 1972, 29:12160			

39	STEAM GENERATORS/NONDESTRUCTIVE TESTING Fuels and Materials Development Program quarterly progress report for period ending March 31, 1973, 29:966 (ORNL-TM-4250)	米		
40	Methods of nondestructive testing during the fabrication and in-service inspection of concrete vessels, of their steel liner, and of the integrated exchangers, 29:15518 (IAEA-145)	仏		
41	STEAM GENERATORS/OPERATION Change in heat-resistance properties of 12Kh1MF steel during use, 29:12474 (ORNL-tr-2724)	ソ		
42	STEAM GENERATORS/PERFORMANCE Discussion of waterside conditions affecting the thermal performance of sodium heated recirculating and once-through steam generators, 29:12161	英	Na	
43	Helical boilers for nuclear systems (AGR, HTGR, PWR, and LMFBR), 29:26006			AGR,HTGR,PWR LMFBR
44	Improvements in tubular heat exchangers (Patent), 29:18295			
45	Once-through steam generators on the line, 29:5033	米		
46	Standards for the calculation of the stability of reactor elements, steam generators, vessels, and piping of atomic power stations, test and research reactors, and installations (Book: in Russian), 29:26269	ソ		
4	Vertical or horizontal steam generators in nuclear power plants with pressurized water reactors, 29:14603	西独	水	PWR
35	STEAM GENERATORS/PERFORMANCE TESTING PSC steam generator hot flow test (HTGR), 29:14611 (GULF-GA-A-12764)	米	He	HTGR
47	STEAM GENERATORS/PIPES Investigations on steam generator tube samples of Inconel 600 after long term service in the FDR, 29:10645 (GKSS-73-E-10)			
48	STEAM GENERATORS/RELIABILITY Steam generators: critical components in nuclear steam supply systems, 29:20265 (AECL-4606)	カ		
49	STEAM GENERATORS/SOLID WASTES Closing the refuse power cycle, 29:23785	米		
50	STEAM GENERATORS/SPECIFICATIONS Evolution of the design of steam generators and intermediate heat exchangers between the Phenix power plant and the design of the 1200 MW(e) power plant, 29:26126	仏	Na	Phenix (FBR)
51	Helical boilers for nuclear systems (AGR, HTGR, PWR, and LMFBR), 29:26006			AGR,HTGR,PWR, LMFBR
52	Large HTGR steam generator design verification and support programs. Heat transfer, fluid flow, and stability, 29:28640 (GULF-GA-A-12808(Vol.2))	米	He	HTGR
53	Steam cycle analysis for modified 300 MW(e) CANDU type reactor, 29:23470	カ	重水	CANDU (PHWR)
54	STEAM GENERATORS/STRESS ANALYSIS Evaluation of steam generator tube, tube sheet and divider plate under combined LOCA plus SSE conditions, 29:23591 (WCAP-7832)	米		
55	STEAM GENERATORS/STRESS CORROSION Fuels and Materials Development Program quarterly progress report for period ending June 30, 1973, 29:5436 (ORNL-TM-4355)	米		

56	STEAM GENERATORS/SUPPORTS Steam generator cradle support (US patent), 29:17721			
57	STEAM GENERATORS/TEMPERATURE DISTRIBUTION Temperature regime of a steam-generating surface on departure from nucleate boiling (DNB), 29:15496			
58	STEAM GENERATORS/TENSILE PROPERTIES Testing and checking of hot-steam components, 29:2594	西独		
59	STEAM GENERATORS/TESTING Experiences acquired from the damage to the reheater in the 50 MW test facility at Hengelc, Holland, 29:29222 (TID-26487)	米		
52	Large HTGR steam generator design verification and support programs. Heat transfer, fluid flow, and stability, 29:28640 (GULF-GA-A-12808(Vol.2))	米	He	HTGR
13	Large HTGR steam generator design verification and support programs. Summary, 29:28639 (GULF-GA-A-12808(Vol.1))	米	He	HTGR
26	STEAM GENERATORS/THERMAL STRESSES Gas-cooled reactor programs annual progress report for period ending December 31, 1972, 29:23442 (ORNL-4911)	米		GCR
61	Thermal stresses in fast reactor boiler tubes at the sodium/gas interface, 29:11870 (CONF-730942-(P2))	英	Na	
18	STEAM GENERATORS/TRANSIENTS Method of characteristics for pressure transients in cylindrical geometry with axial symmetry: the formulation (LMFBR), 29:17571 (ANL-CT-73-23)	米		LMFBR
62	STEAM GENERATORS/TRANSITION BOILING Dynamics of two-phase layer in the transition zone with bubbling, 29:10083	ソ		
11	STEAM GENERATORS/TUBES Experimental studies on flow induced vibration to support steam generator design. Part II. Tube vibration induced by liquid cross-flow in the entrance region of a steam generator, 29:2592 (AECL-4515)	カ		
12	Experimental studies on flow induced vibration to support steam generator design. Part I. Vibration of a heated cylinder in two-phase axial flow, 29:2591 (AECL-4514)	カ		
63	Explosive plugging for sealing tube leaks, 29:11854			

1974 Vol. 30

74000	STEAM GENERATORS			
64	Dynamics of steam generators of a VVER-type nuclear power plant, 30:5375 (EGU-5120208)	チェコ		VVER
65	Power generating system (Patent; HTGR), 30:1904			HTGR
66	Power plant and method for generating peak therein (Patent), 30:20483			
67	Steam generators for the 3800-MW(th) Westinghouse PWR plant, 30:17606	米	水	PWR
68	STEAM GENERATORS/BURNOUT Burn-out correlation to scale water with freon, 30:20983 (RT/ING-(74)7)	伊		
69	Burn-out correlation for once-through L.M.F.B.R. steam generators, 30:23105 (RT/ING-(74)8)	伊	Na	LMFBR
68	Burn-out correlation to scale water with Freon, 30:23473 (RT/ING-(74)7)	伊		
70	STEAM GENERATORS/CONTROL EQUIPMENT Stop shrink and swell oscillations in the nuclear power plant. Upgrade safety and extend life (PWR), 30:20344	米	水	PWR
71	STEAM GENERATORS/CONTROL SYSTEMS Commissioning experience with the control systems for the French nuclear power stations EDF-4 and EL-4, and correlations with results of mathematical simulations, 30:25853	仏	CO_2	EDF-4(GCR) EL-4 (GCR)
72	Hybrid simulation of the steam generating system of a LMFBR with control loops, 30:25883	オランダ		LMFBR
73	Improvement of control behavior of steam generators by structure-optimized control, 30:8883	西独		
74	Operational and fault transient studies of a once-through AGR boiler using the OSCA digital computer program, 30:25849	英	CO_2	AGR
75	Requirements for nuclear boilers, their operating constraints, and dynamic performance, 30:26028	英		
76	STEAM GENERATORS/CORROSION RESISTANCE Corrosion behavior of alloy 600 in nuclear once-through steam generator service, 30:16402	米		
77	STEAM GENERATORS/DESIGN Assessment of actual loading condition of steam generators and heat exchangers according to probabilistic criteria, 30:8838 (CONF-730942-P5)	オーストリア		
78	Code for predicting the thermal performance of a once-through straight shell-and-tube sodium-heated generator, 30:20976 (EUR-4610)	EC	Na	
79	New construction for heat exchangers and steam generators, 30:31899 (AED-CONF-73-223-001)	オーストリア		
80	Once-through vapor generator (Patent), 30:29278			
81	STEAM GENERATORS/DRYOUT Experimental study on heat transfer in long, sub-critical once-through steam generators (LMFBR), 30:11879	伊		LMFBR
82	STEAM GENERATORS/FABRICATION Analysis of some aspects for fast sodium cooled reactor steam generator development, 30:17738	チェコ	Na	LMFBR
83	Fuels and Materials Development Program quarterly progress report for period ending March 31, 1974, 30:24596 (ORNL-TM-4620)	米		LMFBR
84	LMFBR sodium heated steam generator: test model, 30:25864 (CENC-1197(Vol.2))	米	Na	LMFBR

85	STEAM GENERATORS/FAILURES Post-test examination of Alco/BLH sodium-heated steam generator, 30:14544 (ORNL-4950)	米	Na	
86	Steam generator tube failures: world experience in water-cooled nuclear power reactors during 1972, 30:8632 (AECL-4753)	カ	水	LWR
87	STEAM GENERATORS/HEAT TRANSFER Calculating the process of unsteady heat transfer in steam generators with a single-phase incompressible heat carrier with allowance for the distribution of parameters over the length, 30:1854	ソ	水	LWR
88	Experimental study on heat transfer in long, sub-critical once-through steam generators (LMFBR), 30:11879	伊		LMFBR
88	Experimental study on heat transfer in long sub-critical once-through steam generators, 30:25876 (RT/ING-(74)2)	伊		LMFBR
89	Film thickness and flow boiling for two-phase annular flow in helically coiled tubes, 30:12012	日	水－蒸気 air-water	
90	Heat transfer regimes and transitions in boiler dynamics models (Liquid metal cooled reactors), 30:25884	英	液体金属	
91	Heat transfer in once-through steam generators, 30:26709	伊		LMFBR
92	LMFBR demonstration plant steam generating system, 30:28681 (CENC-1194(Vol.4)(Suppl.1))	米		LMFBR
93	Two-phase flow experiments in a 10 m long sodium heated steam generator test section (LMFBR), 30:1880	オランダ	Na	LMFBR
94	STEAM GENERATORS/INSPECTION Disassembly and examination of SRE sodium heated steam generator. Reactor 7978, 30:17695 (NP-20046)	米	Na	SRE
95	STEAM GENERATORS/LEAKS Problems with sodium steam generators: example of a leakage in a steam generator of the sodium-cooled KNK nuclear power station, 30:34249	西独	Na	KNK
96	STEAM GENERATORS/MATHEMATICAL MODELS Boiler simulation, PWR station dynamics, 30:25832	英	水	PWR
97	Comparison of the boiler model NUBOIL with transient measurements on a once-through boiler rig (Gas cooled reactors), 30:25852	英		GCR
98	Dynamic behaviour of a 660 MW nuclear power station, 30:25833	西独		
99	Experimental and analytical modelling in boiler design, 30:26031	英		
72	Hybrid simulation of the steam generating system of a LMFBR with control loops, 30:25883	オランダ		LMFBR
100	Modelling of once-through boiler dynamics (Gas cooled reactors), 30:25848	英	CO_2	AGR,HTR
101	STEAM GENERATORS/MEETINGS Boiler dynamics and control in nuclear power stations. Proceedings of international conference held in London, England, March 22--23, 1973, 30:26027			
102	STEAM GENERATORS/NONDESTRUCTIVE TESTING Fuels and material development program. Quarterly progress report for period ending March 31, 1972, 30:21665 (ORNL-TM-3797)	米		
83	Fuels and Materials Development Program quarterly progress report for period ending March 31, 1974, 30:24596 (ORNL-TM-4620)	米		LMFBR

103	Recent advances in NDT of steam generators for LMFBR, 30:23136	米		LMFBR
78	STEAM GENERATORS/PERFORMANCE Code for predicting the thermal performance of a once-through straight shell-and-tube sodium-heated generator, 30:20976 (EUR-4610)	EC	Na	
75	Requirements for nuclear boilers, their operating constraints, and dynamic performance, 30:26028	英		
104	Steam generators in nuclear power stations, 30:23207	Na		
105	STEAM GENERATORS/PERFORMANCE TESTING LMFBR modular steam generator test results, 30:1958	伊		LMFBR
106	Steam generator test loop for sodium cooled fast breeder reactor, 30:20381	米	Na	LMFBR
107	Tests on the CGVS of the 45 MW mock-up of Phenix steam generator, 30:1944	日	Na	Phenix(FBR)
108	Thermal-hydraulic performance of a 2MW/sub t/ sodium-heated, forced recirculation steam generator model, 30:20386	仏	Na	
109	STEAM GENERATORS/POOL BOILING Boiling curve measurements from a controlled forced convection process (BWR), 30:25821	米	水	BWR
110	STEAM GENERATORS/RELIABILITY Steam generator reliability: the Canadian approach, 30:8678 (AECL-4771)	カ		
111	Steam generator reliability: the Canadian approach (PHWR), 30:34218	カ	重水	PHWR
112	STEAM GENERATORS/RESEARCH PROGRAMS Large HTGR steam generator design verification and support programs: materials, 30:25839 (GULF-GA-A-12808(Vol.3))	米	He	HTGR
113	LMFBR demonstration plant steam generating system, 30:31217 (CENC-1194(Vol.1)(Suppl.1))	米	Na	LMFBR
114	LMFBR demonstration plant steam generating system, 30:31218 (CENC-1194(Vol.2)(Suppl.1))	米	Na	LMFBR
115	LMFBR sodium heated steam generator: test model, 30:31219 (CENC-1197(Vol.1))	米	Na	LMFBR
116	LMFBR sodium heated steam generator: test model, 30:31220 (CENC-1197(Vol.3))	米	Na	LMFBR
117	LMFBR sodium heated steam generator: test model (Structural steel analysis), 30:31221 (CENC-1197(Vol.4))	米	Na	LMFBR
118	Research and development of steam generators for pressurized water reactors, 30:23076	伊	水	PWR
82	STEAM GENERATORS/SPECIFICATIONS Analysis of some aspects for fast sodium cooled reactor steam generator development, 30:17738	チェコ	Na	LMFBR
119	Design of pressure vessels for elevated temperature service, 30:12009	米		
120	LMFBR sodium heated steam generator: test model, 30:25864 (CENC-1197(Vol.2))	米	evaporator superheater	Na LMFBR
121	STEAM GENERATORS/STANDARDS Requirements for nuclear components. (Supplement to ASME boiler and pressure vessel code, section III), 30:12005 (RDT-E-15-2-T(5-74))	米		
122	Sodium-heated steam generator (LMFBR), 30:1936 (RDT-E-4-16-T(2-74))	米	Na	LMFBR
123	STEAM GENERATORS/STRESS ANALYSIS Assessment of actual loading conditions of steam generators and heat exchangers according to probabilistic criteria, 30:8875	オーストリア		

124	Example of the evaluation of fatigue strength of a high temperature structure by ASME code, 30:17890	米		
125	STEAM GENERATORS/TRSTING Liquid Metal Engineering Center (LMEC) annual technical progress report, January--December 1973 (LMFBR sodium component and instrumentation testing), 30:1934 (LMEC-74-1)	米	Na	LMFBR
126	Post-test examination of Alco/BLH sodium-heated steam generator, 30:14544 (ORNL-4950)	米	Na	
127	STEAM GENERATORS/THERMAL STRESSES Design of pressure vessels for elevated temperature service, 30:12009	米		
128	STEAM GENERATORS/TRANSIENTS Transient behaviour of nuclear steam generators, 30:14592 (INIS-mf-1323)	オランダ		LMFBR PWR
129	STEAM GENERATORS/TUBES Corrosion of steam generator tubing in operating pressurized water reactors, 30:34161 (BNL-19158)	米	水	PWR
130	Inspection feasibility of grooved duplex tubes (LMFBR), 30:11849 (ANL-8074)	米	Na	LMFBR
89	STEAM GENERATORS/TWO-PHASE FLOW Film thickness and flow boiling for two-phase annular flow in helically coiled tubes, 30:12012	日	air-水 水-蒸気	
131	LMFBR demonstration plant steam generating system, 30:28681 (CENC-1194(Vol.4)(Suppl.1)	米	Na	LMFBR
132	Review of two phase flow instability aspects of boiler dynamics, 30:26030	英		
133	Two-phase flow experiments in a 10 m long sodium heated steam generator test section (LMFBR), 30:11880	オランダ	Na	LMFBR
134	STEAM GENERATORS/UNSTEADY FLOW Flow stability and dynamic behavior of nuclear boilers (HTGR), 30:25851	オーストリア		HTR
135	Study of oscillatory instabilities in the parallel channels of a high pressure once-through boiler rig (AGR), 30:25850	英	CO_2	AGR

1975 Vol. 31

75000	STEAM GENERATORS			
1	Deposition kinetics of calcium hydroxy apatite on heat transfer surfaces at boiling, 31:3218 (AE-498)	スウェーデン		
2	Heat exchanger (Patent), 31:25014			
3	Steam-generator installation equipment (HTGR), 31:24632 (TID/SNA-177)	米	He	HTGR
4	STEAM GENERATORS/AVAILABILITY Estimate of LMFBR steam generator system reliability and availability, 31:2369 (CONF-740401-P3)	米		LMFBR
5	STEAM GENERATORS/BUILDING MATERIALS Stainless steel and nickel alloy tubes for nuclear power generation (Steam generators and fuel cans for PWR, HTR, LMFBR), 31:17117	スウェーデン		PWR,HTR LMFBR
6	STEAM GENERATORS/BURNOUT Direct and indirect heating effect in the DNB tests for sodium heated steam generators for LMFBR's, 31:15631 (RT/ING-(74)22)	伊	Na	LMFBR
7	STEAM GENERATORS/CORROSION Corrosion phenomena following a sodium--water microreaction in a crack crossing the wall of a fast reactor steam generator tube (Steel 10 CD 9-10; anstenitic steel Z6CNT 18-10), 31:1108	仏	Na	
8	ETS-1 steam generator injector face weld corrosion (NERVA program), 31:22409 (RN-TM-0370)	米		NERVA
9	STEAM GENERATORS/CORROSION RESISTANCE Steam generator materials engineering first quarterly report, July--September 1974 (LMFBR; $2^1/_4$Cr-1Mo and stainless steels), 31:30153 (GEAP-14029-1)	米		LMFBR
10	STEAM GENERATORS/DESIGN Design and fabrication of pressurized water nuclear steam supply systems in France, 31:27965	仏	水	PWR
11	Italian programme of sodium-heated steam generators, 31:13202	伊	Na	LMFBR
12	Nuclear steam generator (Patent), 31:5332			
13	STEAM GENERATORS/DOCUMENT TYPES Information for safety analysis reports: steam generators (PWR), 31:25096 (REG/G-1.70.19(01-75))	米	水	PWR
6	STEAM GENERATORS/DRYOUT Direct and indirect heating effect in the DNB tests for sodium heated steam generators for LMFBR's, 31:15631 (RT/ING-(74)22)	伊	Na	LMFBR
14	Dryout in helically coiled tube of sodium heated steam generator (LMFBR), 31:31845	日	Na	LMFBR
15	STEAM GENERATORS/EROSION Preliminary correlation of sodium/water wastage data and its application (LMFBR), 31:7635	米	Na	LMFBR
16	STEAM GENERATORS/EXPLOSION WELDING Remote maintenance of nuclear steam generators, 31:35607	西独		
17	STEAM GENERATORS/FABRICATION Construction, fabrication and assembly of steam generators, valves and pipings in nuclear power stations. P. 1, 31:13156	西独	水	Biblis (PWR)
18	Construction, fabrication and assembly of steam generators, valves and pipings in nuclear power stations. P. 2, 31:13157	西独		
19	Construction, fabrication and assembly of steam generators, valves and pipings in nuclear power stations. P. 3, 31:13158	西独		

20	Design and fabrication of pressurized water nuclear steam supply systems in France, 31:27965	仏	水	PWR
21	Steam pressure systems. Main primary circuit of boiling-water reactor. 1. Decision of February 26, 1974 applying regulations on pressure systems to boiling-water reactors. 2. Circular of February 26, 1974 on the application of regulations for steam pressure systems to the main primary circuit of boiling-water reactors, 31:24588 (NP-tr-1953)		水	
22	STEAM GENERATORS/FAILURES Bubble formation and growth (LMFBR), 31:5261 (ANL-CT-73-15)	米		LMFBR
23	Computational methods of evaluating the emergency state of a sodium loop during piping breakdown in a "sodium--water" steam generator (LMFBR), 31:19038 (CONF-701074-P1)	ソ	Na-水	LMFBR
24	Plastic wave propagation in an axially symmetric circular cylindrical shell (LMFBR), 31:28001 (ANL-CT-75-27)	米		LMFBR
25	Steam generator tube failures: world experience in water-cooled nuclear power reactors in 1973 (BWR and PWR), 31:31745 (AECL-5013)	力	水	BWR,PWR
26	Thermal sleeve failure and repairs-Indian Point No. 1 nuclear unit (285 MW), 31:13159	米	水	Indian Point 1 (PWR)
27	TRANSWRAP: code for analyzing the system effects of large-leak sodium--water reactions in LMFBR steam generators, 31:2433 (CONF-740401-P1)	米	Na-水	LMFBR
28	STEAM GENERATORS/GAS FLOW Heat transfer on coiled tubular matrix, 31:31783	米		
29	STEAM GENERATORS/HEAT TRANSFER Boiling heat transfer in LMFBR steam generators, 31:2378 (RT/ING-(74)17)	伊	Na	LMFBR
14	Dryout in helically coiled tube of sodium heated steam generator (LMFBR), 31:31845	日	Na	LMFBR
30	Heat transfer to dry and superheated steam in once-through generators for LMFBR's, 31:2379 (RT/ING-(74)18)	伊	Na	LMFBR
28	Heat transfer on coiled tubular matrix, 31:31783	米		
31	STEAM GENERATORS/LEAK DETECTORS Acoustic detection for small-leak sodium--water reaction (LMFBR), 31:2356 (CONF-740401-P1)	日	Na-水	LMFBR
32	Use of in-sodium hydrogen detectors during steam generator tests in the SCTI (LMFBR), 31:2355 (CONF-740401-P1)	米	Na	LMFBR
33	STEAM GENERATORS/LEAK TESTING Evaluation of steam-to-sodium leak detectors in the sodium components test installation (SCTI), 31:14223 (CONF-741085-1)	米	Na-水	LMFBR
34	STEAM GENERATORS/LEAKS Behavior of small leaks in LMFBR steam generators, 31:7616	米	Na	LMFBR
35	Steam generator failure at the KNK-reactor plant, 31:30149 (AED-CONF-74-538-015)	西独	Na	KNK
36	STEAM GENERATORS/LMFBR TYPE REACTORS Stress corrosion studies of LMFBR steam generator materials (9Cr--1Mo; Carpenter 20Cb-3; Armco 22-13-5; Uranus-50: $2\frac{1}{4}$Cr--1Mo), 31:30152 (GEAP-12536)	米	Na	LMFBR
37	STEAM GENERATORS/MATERIALS TESTING Investigation of the behaviour of tube materials for steam generators of pressurized water reactors (Corrosion esistance), 31:24607 (BMFT-FB-K-74-03)	西独	水	PWR

38	Materials selection and expected performance in near term LMFBR steam generators, 31:13206	米	Na	LMFBR
39	STEAM GENERATORS/MECHANICAL VIBRATIONS Large HTGR steam generator design verification and support programs: vibration, 31:5190 (GA-A-12808(Vol.4))	米	He	HTGR
40	STEAM GENERATORS/MOCKUP Dept. operation of 50 MW test loop. Report concerning weeks 1 to 5, 1974 (Sodium cooled reactors), 31:10362 (TNO-1)	オランダ	Na	
41	STEAM GENERATORS/PERFORMANCE Hydrodynamic stability and thermal performance test of a 1-MWt sodium-heated once-through steam generator model, 31:25628	米	Na	
42	Hypercritical once-through vapor generators in counter flow with and without twisted tapes, 31:3222	伊		
43	Interaction of steam generator modules during a sodium--water reaction (LMFBR), 31:2431 (CONF-740401-P1)	米	Na-水	LMFBR
44	Operating experience with steam generators in gas-cooled reactors, 31:7546	米		GCR
40	STEAM GENERATORS/PERFORMANCE TESTING Dept. operation of 50 MW test loop. Report concerning weeks 1 to 5, 1974 (Sodium cooled reactors), 31:10362 (TNO-1)	米	Na	
45	Dept. operation of 50 MW test loop. Report concerning weeks 6 to 9, 1974 (Sodium cooled reactors), 31:10363 (TNO-2)	米	Na	
8	ETS-1 steam generator system improvement test program (NERVA Program), 31:29359 (NTO-R-0125)	米		NERVA
47	Steam generator development test program, Experimental Plan V. Test report (NERVA Program), 31:29354 (NTO-R-0082)	米		NERVA
48	STEAM GENERATORS/PLANNING Development of sodium-heated steam generators in the Netherlands, 31:13205	オランダ	Na	
49	STEAM GENERATORS/PLATES Thermal stresses on the surface of perforated plates with temperature distributions varying through the thickness only, 31:19472 (WAPD-TM-1190)	米		
50	STEAM GENERATORS/PRESSURE RELEASE Dynamic transient analysis of rupture disks (LMFBR), 31:35166 (ANL-CT-75-33)	米		LMFBR
24	Plastic wave propagation in an axially symmetric circular cylindrical shell (LMFBR), 31:28001 (ANL-CT-75-27)	米		LMFBR
51	STEAM GENERATORS/RELIABILITY Estimate of LMFBR steam generator system reliability and availability, 31:2369 (CONF-740401-P3)	米		LMFBR
52	STEAM GENERATORS/REMOTE CONTROL ETS-1 console operations manual, 31:32817 (NTO-T-0003)	米		
53	STEAM GENERATORS/REMOTE HANDLING Volume I. System analysis. Steam generator plenum service system. Phase I. Preliminary design. Final summary report to Gulf General Atomic, Inc. (HTGR), 31:24625 (NDS-5(Vol.1))	米	He	HTGR
54	Volume V. Equipment operation and engineering analysis. Steam generator plenum service system. Phase 1. Preliminary design. Final summary report to Gulf General Atomic, Inc. (HTGR), 31:24629 (NDS-5(Vol.5)(Phase 1))	米	He	HTGR

55	STEAM GENERATORS/REMOTE HANDLING EQUIPMENT Volume II. Equipment design criteria and descriptions. Steam generator plenum service system. Phase I. Preliminary design. Final summary report to Gulf General Atomic, Inc. (HTGR), 31:24626 (NDS-5(Vol.2)(Ph.1))	米	He	HTGR
56	Volume VI. Program planning, Phases II through V. Steam generator plenum service system. Phase I. Preliminary design. Final summary report to Gulf General Atomic, Inc. (HTGR), 31:24630 (NDS-5(Vol.6))	米	He	HTGR
57	STEAM GENERATORS/REMOTE VIEWING EQUIPMENT Proposal No. 681023 to Gulf General Atomic Inc. for steam generator plenum viewing system (Fort St. Vrain Reactor), 31:24631 (TID/SNA-174)	米	He	Fort St. Vrain (HTGR)
58	STEAM GENERATORS/REPAIR Remote maintenance of nuclear steam generators, 31:32247 (AED-CONF-74-538-027)	米		
59	Repairs of discrepant steam generating system, liquid lines (NERVA program), 31:22392 (RN-DR-0083)	米		NERVA
60	Thermal sleeve failure and repairs-Indian Point No. 1 nuclear unit (285 MW), 31:13159	米	水	Indian Point 1 (PWR)
50	STEAM GENERATORS/SAFETY ENGINEERING Dynamic transient analysis of rupture disks (LMFBR), 31:35166 (ANL-CT-75-33)	米	Na	LMFBR
61	STEAM GENERATORS/SPECIFICATIONS Assessment of actual loading conditions of steam generators and heat exchangers according to probabilistic criteria, 31:5321	オーストリア		
48	Development of sodium-heated steam generators in the Netherlands, 31:13205	オランダ	Na	
49	High-power steam generators for atomic electric power plants with water-moderated water-cooled reactors (PWR), 31:18744 (CONF-701074-P1)	ソ	水	PWR
62	LMFBR steam generator material selection, development program, and test results ($2^1/_4$ Cr--1 Mo steel), 31:13189 (CONF-741114-1)	米		LMFBR
63	Performance design and product confirmation requirements for steam generation and delivery system (NERVA Altitude Simulation System), 31:25540 (TID/SNA-119)	米		NERVA
64	Safety considerations in the design and constructions of nuclear steam generators, 31:21840	オーストリア		
65	Single-pass steam generator with sodium coolant (LMFBR), 31:18773 (CONF-701074-P1)	ソ	Na	LMFBR
66	Specifications for ETS-1 steam generator system (NERVA Program), 31:29387 (TID/SNA-1239)	米		NERVA
67	STEAM GENERATOTS/TESTING Steam generator system development test report EP-IV, EP-V, EP-VI (NERVA), 31:28244 (RN-S-0360)	米		NERVA
68	STEAM GENERATORS/THERMAL STRESSES DNB induced thermal stress and fatigue in LMFBR evaporator tubes with oxide scale, 31:24649 (ANL-CT-75-24)	米		LMFBR
69	STEAM GENERATORS/THERMODYNAMIC CYCLES Method and apparatus for regenerative heating in thermal power plants (Patent), 31:13371			
70	STEAM GENERATORS/TUBES 2-1/4-percent-chromium, 1-percent-molybdenum alloy steel seamless tubes (ASME SA-213 with additional requirements), 31:30169 (RDT-M-3-33T(2-75))	米		

71	Preliminary results from small leak tube wastage scoping tests (LMFBR), 31:2354 (CONF-740401-P1)	米	Na	LMFBR
72	Wastage of steam generator tubes during small leak of steam into sodium (LMFBR), 31:2353 (CONF-740401-P1)	日		LMFBR
73	STEAM GENERATORS/TWO-PHASE FLOW Waterside flow problems in boilers, 31:25008	英	水	
74	STEAM GENERATORS/VENTURI TUBES ETS-1 generator Experimental Plan VIII. Venturi calibration test program, 31:29356 (NTO-R-0101)	米		
75	STEAM GENERATORS/WELDED JOINTS Some welding requirements of the CANDU nuclear steam supply system, 31:7551	カ	重水	CANDU (PHWR)

1975 Vol. 32

75000	STEAM GENERATORS			
76	6-MW electric steam generator for startup tests of nuclear plants, 32:2816			
77	A heat-exchanger apparatus, specially suitable as steam generator (Patent), 32:16625			
78	Dynamic simulation of an integral economizer, U-tube steam generator (PWR), 32:13076	米	水	PWR
79	Investigations on HTR-heat exchanger materials, 32:2416 (JUL-1111-RG)	西独		HTR
80	Liquid metal steam generator (Patent: LMFBR), 32:25292		液体金属	LMFBR
81	STEAM GENERATORS/BUILDING MATERIALS Comparative results on chloride stress corrosion cracking of steam generator materials in cyclic steam environment, 32:14797 (ORNL-5031)	米		
82	STEAM GENERATORS/BURNOUT Burnout power in once-through tubular steam generators (LMFBR power plants), 32:19641	伊	Na	LMFBR
83	STEAM GENERATORS/COMPARATIVE EVALUATIONS The safety in steam generator operation in pressurized water reactors, 32:16223	ソ	水	PWR
84	STEAM GENERATORS/CORROSION Effects of service environments on the behavior of HTGR steam generator structural materials, 32:25250 (GA-A-13553)	米	He	HTGR
85	Importance of thermohydraulic conditions for corrosion safety of PWR-steam generators, 32:16224	スイス		PWR
86	Significance of thermohydraulic conditions for the corrosion safety of PWR steam generators, 32:23574 (LIB/Trans-567)		水	PWR
87	Studies of a mechanism for material wastage by sodium-water reaction jets, 32:3939	英		
88	STEAM GENERATORS/CORROSION PROTECTION Water chemistry and corrosion limitation at the BBR steam generator system (PWR), 32:10742	西独	水	PWR
89	Water chemistry and corrosion limitation in the BBR-steam generator system (PWR), 32:16222	西独	水	PWR
90	STEAM GENERATORS/CRITICAL HEAT FLUX Critical heat flux for large diameter steam generating tubes with circumferentially variable and uniform heating, 32:19638	英		
91	Heat transfer crisis at the steam water flowing in channels in high steam content zones (Thecretical and experimental studies), 32:19642	ソ		
92	STEAM GENERATORS/DECONTAMINATION Cleaning contaminated surfaces of nuclear power plants and recovery of removed surface particles (Patent; water cooled and heavy water cooled reactors), 32:25461		水	LWR HWR
93	STEAM GENERATORS/DESIGN Clean power: for the nuclear future (Foster Wheeler Corporation activities), 32:23992	米		HTGR PBR LWR
94	Design guide for heat transfer equipment in water-cooled nuclear reactor systems, 32:18802 (ORNL-TM-3578)	米	水	LWR
95	In the thermal design of steam generators for LMFBRs, 32:2360	伊	Na	LMFBR

96	Trip report: United States LMFBR Steam Generator Team. IAEA symposium, Bensberg, Germany, October 14--17, 1974, 32:25267 (ERDA-36)	米	Na	LMFBR
97	STEAM GENERATORS/DRYOUT Assesment of boiler tube temperature change during dryout in nuclear steam raising plant (Sodium cooled reactor), 32:21741	英	Na	
98	Boiling 6: crisis (Calculation and measurement techniques and theoretical studies), 32:19655	ソ		
99	STEAM GENERATORS/EFFICIENCY Steam power plant with a pressurized-water reactor and several steam generators heated with pressurized water (Patent), 32:30154		水	PWR
93	STEAM GENERATORS/FABRICATION Clean power: for the nuclear future (Foster Wheeler Corporation activities), 32:23992	米		HTGR FGR LWR
100	Directions of steam-generator progress for pressure water reactors, 32:5342	チェコ	水	PWR
96	Trip report: United States LMFBR Steam Generator Team, IAEA symposium, Bensberg, Germany, October 14--17, 1974, 32:25267 (ERDA-36)	米	Na	LMFBR
101	STEAM GENERATORS/FAILURES Steam generators: the long road to maturity, 32:2191	カ		
102	Steam-generator tube failures: world experience in water-cooled nuclear power reactors in 1973, 32:30140	カ	水	LWR
88	STEAM GENERATORS/FEEDWATER Water chemistry and corrosion limitation at the BBR steam generator system (PWR), 32:10742	西独	水	PWR
103	STEAM GENERATORS/FLUID FLOW Large HTGR steam generator design verification and support programs, 32:5355 (GA-A-12808)	米	He	HTGR
104	Stability behaviour of steam generators with upward flow of helium. Final report. Dragon Project report (Dragon), 32:27237 (DP-Report-625)	英	He	Dragon (HTGR)
105	WOTAN: a digital program for the simulation of a once-through steam generator, 32:21723 (TUBIK-40)	西独		
106	STEAM GENERATORS/HEAT TRANSFER Boiling heat transfer in LMFBR steam generators, 32:5395 (RT/ING-(74)17)	伊	Na	LMFBR
107	Determining the heat transfer on tube bundle support plates, 32:11439			
108	Evaluation of piping heat transfer, piping flow regimes, and steam generator heat tranasfer for the Semiscale Mod-1 isothermal tests, 32:27365 (ANCR-1229)	米		
109	Heat transfer in sub-critical once-through steam generators, 32:11084 (RT/ING-(74)32)	伊		
103	Large HTGR steam generator design verification and support programs, 32:5355 (GA-A-12808)	米		HTGR
110	Steam generator installations for nuclear power stations (Book), 32:13024 (ERDA-tr-7)	ソ		
105	WOTAN: a digital program for the simulation of a once-through steam generator, 32:21723 (TUBIK-40)	西独		
108	STEAM GENERATORS/HYDRAULICS Evaluation of piping heat transfer, piping flow regimes, and steam generator heat transfer for the Semiscale Mod-1 isothermal tests, 32:27365 (ANCR-1229)	米		

110	STEAM GENERATORS/HYDRODYNAMICS Steam generator installations for nuclear power stations (Book), 32:13024 (ERDA-tr-7)	ソ		
111	STEAM GENERATORS/INSPECTION Inservice inspection of pressurized water reactor steam generator tubes, 32:24060 (REG/G-1.83(Rev.1)(7-75))	米	水	PWR
112	STEAM GENERATORS/LEAK TESTING Leak understood (Steam generator tubes), 32:5698	英		
113	Modern leak tests of steam generators, 32:14158	西独		
114	STEAM GENERATORS/LEAKS Leak detection in fast neutron reactor steam generators, 32:10777 (ERDA-tr-26)	ソ	Na	BN-350
115	TRANSWRAP: a compressible hydrodynamic code for large-leak sodium/water reaction analysis, 32:5397 (TI-001-130-025)	米	Na	
116	STEAM GENERATORS/MAINTENANCE 50 MWe test loop. Monthly report, July--August 1974 (weeks 27--35) (Sodium cooled reactors), 32:30413 (TNO-LMEC-7)	オランダ		
117	Chemical steam generating system. Operation and maintenance manual (NERVA Program), 32:400 (TID/SNA-1315)	米		NERVA
118	STEAM GENERATORS/MATERIALS U.S. Advanced Materials Development Program for steam generators (LMFBR), 32:25265 (CONF-750911-2)	米	Na	LMFBR
119	STEAM GENERATORS/MECHANICAL PROPERTIES Fuels and materials development program quarterly progress report for period ending March 31, 1975 (Advanced absorber materials; steam generator materials; mechanical properties; weldments for LMFBR), 32:28478 (ORNL-TM-4940)	米		LMFBR
120	STEAM GENERATORS/MOCKUP 50 MWe test loop. Monthly report, June 1974 (weeks 23--26) (Sodium coded reactors), 32:30412 (TNO-LMEC-6)	オランダ	Na	
121	50 MWe test loop. Monthly report, July--August 1974 (weeks 27--35) (Sodium cooled reactors), 32:30413 (TNO-LMEC-7)	オランダ	Na	
122	50 MWe test loop. Monthly report, September 1974 (weeks 36--39) (Sodium cooled reactors), 32:30414 (TNO-LMEC-8)	オランダ	Na	
123	Department for operation of 50 MW test loop. Monthly report, March 1974 (weeks 10--13) (Sodium cooled reactor mockup), 32:5695 (TNO-LMEC-3)	オランダ	Na	
124	Department for operation of 50 MW test loop. Monthly report, April 1974 (weeks 14--17) (Sodium cooled reactor mockup), 32:5696 (TNO-LMEC-4)	オランダ	Na	
125	Department for operation of 50 MW test loop. Monthly report, May 1974 (weeks 18--22) (Sodium cooled reactor mockup), 32:5697 (TNO-LMEC-5)	オランダ	Na	
119	STEAM GENERATORS/NONDESTRUCTIVE TESTING Fuels and materials development program quarterly progress report for period ending March 31, 1975 (Advanced absorber materials; steam generator materials; mechanical properties; weldments for LMFBR), 32:28478 (ORNL-TM-4940)	米		LMFBR
117	STEAM GENERATORS/OPERATION Chemical steam generating system. Operation and maintenance manual (NERVA Program), 32:400 (TID/SNA-1315)	米		NERVA
126	STEAM GENERATORS/PERFORMANCE Investigation of heat transfer to potassium in coil tubes and sodium--potassium coil tubes generator (Correlation of heat transfer characteristics of potassium vapor generator and sodium-heated steam generator), 32:19649	ソ	NaK	

127	Sensitivity predictions via parameter variation in the steady state and transient computational models of the Argonne SGTF, 32:21485 (ANL-CT-75-20)	米		SGTF
128	Testing of a bayonet-type integrated steam generator system for sodium-cooled breeders, 32:2366		Na	LMFBR
129	World survey of operating experience with water-cooled nuclear reactor steam generators, 32:30408 (INIS-mf-1521)	カ	水	LWR
120	STEAM GENERATORS/PERFORMANCE TESTING 50 MWe test loop. Monthly report, June 1974 (weeks 23--26) (Sodium coded reactors), 32:30412 (TNO-LMEC-6)	オランダ	Na	
122	50MWe test loop. Monthly report, September 1974 (weeks 36--39) (Sodium cooled reactors), 32:30414 (TNO-LMEC-8)	オランダ	Na	
82	Burnout power in once-through tubular steam generators (LMFBR power plants), 32:19641	伊	Na	LMFBR
123	Department for operation of 50 MW test loop. Monthly report, March 1974 (weeks 10--13) (Sodium cooled reactor mockup), 32:5695 (TNO-LMEC-3)	オランダ	Na	
124	Department for operation of 50 MW test loop. Monthly report, April 1974 (weeks 14--17) (Sodium cooled reactor mockup), 32:5696 (TNO-LMEC-4)	オランダ	Na	
125	Department for operation of 50 MW test loop. Monthly report, May 1974 (weeks 18--22)(Sodium cooled reactor mockup), 32:5697 (TNO-LMEC-5)	オランダ	Na	
130	Sodium/water reaction tests for a protected bayonet tube steam generator (LMFBR), 32:13205	米		LMFBR
131	Test description for Experimental Plan V steam generator development test program (NERVA Program), 32:3293 (NTO-I-0078)	米		NERVA
132	STEAM GENERATORS/PITTING CORROSION Materials performance in operating PWR steam generators, 32:23569 (BNL-20307)	米	水	PWR
119	STEAM GENERATORS/REACTOR MATERIALS Fuels and materials development program quarterly progress report for period ending March 31, 1975 (Advanced absorber materials; steam generator materials; mechanical properties; weldments for LMFBR), 32:28478 (ORNL-TM-4940)	米		LMFBR
133	Steam generator material for sodium-cooled fast breeder reactors: results and future prospects, 32:2351	仏	Na	FBR
134	Steam generator materials engineering second quarterly report, October--December 1974, 32:17748 (GEAP-14029-2)	米		
135	STEAM GENERATORS/RESEARCH PROGRAMS Development of steam generator in liquid metal fast breeder reactor, 32:18855	日	Na	LMFBR
136	STEAM GENERATORS/SHOCK ABSORBERS Energy absorbing structure in a steam generator (Patent), 32:8766			
137	STEAM GENERATORS/SPECIFICATIONS Direct ions of steam-generator progress for pressure water reactors, 32:5342	チェコ	水	PWR
138	STEAM GENERATORS/STRESS CORROSION Comparative results on chloride stress corrosion cracking of steam generator materials in cyclic steam environment, 32:14797 (ORNL-5031)	米		
139	Materials performance in operating PWR steam generators, 32:23569 (BNL-20307)	米	水	PWR

140	STEAM GENERATORS/TEST FACILITIES Computer system for 50 MW steam generator test facility (SGF-50), Power Reactor and Nuclear Fuel Development Corporation, 32:19536	日	Na	
96	STEAM GENERATORS/TESTING Trip report: United States LMFBR Steam Generator Team. IAEA symposium, Bensberg, Germany, October 14--17, 1974, 32:25267 (ERDA-36)	米	Na	LMFBR
141	STEAM GENERATORS/TUBES Critical heat flux in inclined and vertical smooth and ribbed tubes, 32:19640	米		
142	Effect of liquid sodium on the tensile and creep-rupture properties of incoloy-800: a preliminary study (LMFBR), 32:21509 (RD/B/N-3195)	英	Na	LMFBR
143	Joining techniques for remote maintenance of nuclear steam generators, 32:16626	米		
144	Steam-generator tube failures: world experience in water-cooled nuclear power reactors during 1972, 32:8361	カ	水	LWR
145	STEAM GENERATORS/TURBULENT FLOW SS-TVF a computer code for turbulent flow velocity field calculations in square array rod subchannels, 32:19168 (RT/ING-(74)29)	伊		
146	TS-TVF a computer code for turbulent flow velocity field calculation in triangular array rod subchannels, 32:19169 (RT/ING-(74)28)	伊		
147	STEAM GENERATORS/TWO-PHASE FLOW Design correlation for pressure drops in coiled, once-through steam generators, 32:25870 (RT/ING-(75)7)	伊		
148	Dynamic stability of steam--water mixture flow in a once-through steam generator for nuclear power stations, 32:5702	チェコ		
149	STEAM GENERATORS/WATER CHEMISTRY Optimizing the water conditions of the secondary loops of nuclear power stations with water-moderated water-cooled reactors, 32:13067	ソ	水	
150	Possibility of using KU-2 cation resin for polishing condensates at higher temperatures, 32:13027	ソ		
110	Steam generator installations for nuclear power stations (Book), 32:13024 (ERDA-tr-7)	ソ		
151	Steam side water chemistry in PWR steam generators, 32:2240		水	PWR
152	STEAM GENERATORS/WELDED JOINTS Determination of the conditions of ^{170}Tm radiography of tube welds on plate and tube joining welds (Sodium--steam generator), 32:631	仏	Na	
153	Experience in the inductive heat treatment of welded seams and claddings of nuclear reactor components, 32:2807 (AED-CONF-74-538-009)	西独		

1976 Vol. 33

	STEAM GENERATORS				
1	Changes in the conception of the steam generators and the intermediate exchangers between the Phenix plant and the 1200 MWe plant (LMFBR), 33:16285 (ANL-Trans-1009)	仏	IHX,SG	Na	Phenix (FBR)
2	Improvements to steam generators particularly to integrated generators for nuclear plants (Patent; power stations of the fixed type or the marine type), 33:4891				マリン
3	Method and apparatus for reverse circulating nuclear steam generator secondary fluid (Patent), 33:4897				
4	Preliminary studies on dynamics of beam/stop impact: numerical analysis (Tube;baffle vibrational characteristics), 33:2319 (ANL-CT-76-06)	米			
5	Research on thermohydraulic stability of steam generators for light water reactors, 33:27961 (SGAE-2450)	西独		水	LWR
6	Steam generator (Patent; PWR), 33:4647				PWR
7	STEAM GENERATORS/CORROSION Behavior of metals in high-temperature reactor helium for steam generators, 33:27997	英		He	Dragon (HTGR)
8	Compatibility of Fe--$2\frac{1}{4}$ wt % Cr--1 wt % Mo steel in a sodium environment (LMFBR), 33:28044	米		Na	LMFBR
9	STEAM GENERATORS/CORROSION PRODUCTS Accumulation of radioactivecorrosion products in nuclear steam generators, 33:32522	力			
10	STEAM GENERATORS/CORROSION PROTECTION Application of sodium hydrogen phosphate to the once-through steam generator of an English nuclear power station, 33:19594	英			
11	The significance of the iron(III) ions for the corrosion danger in cleaning steam generators, 33:17679	西独			
12	STEAM GENERATORS/DEFORMATION Dynamic plastic deformation of shell: an endochronic solution (LMFBR), 33:22178	米			LMFBR
11	STEAM GENERATORS/DESCALING The significance of the iron(III) ions for the corrosion danger in cleaning steam generators, 33:17679	西独			
13	STEAM GENERATORS/DESIGN Alternate steam generator development program. Phase I summary report, September1974--December 1974. Volume 1. Development of alternatesteam generators for LMFBR plants, 33:22094 (WARD-SG-3045-1(Vol.1))	米		Na	LMFBR
14	Conceptual design of a fluidized bed steam generator for the LMFBR, 33:6752	米		Na	LMFBR
15	Design of fast reactor boilers, 33:11384	英			高速炉
16	Evolution of heat-exchanger design for sodium-cooled reactors, 33:8730	西独	IHX	Na	LMFBR
17	High-temperature gas-cooled reactor steam generator design, 33:27999	米		He	HTGR
18	Influence of secondary failures on LMFBR steam generator system design, 33:22169	米			LMFBR
19	Method and apparatus for generating steam by nuclear explosion with suppressed radiation and blast effects (Patent), 33:20177				
20	Pressurized-water coolant nuclear reactor steam generator (Patent), 33:19441			水	PWR
21	Steam generator for a pressurized-water coolant nuclear reactor (Patent), 33:19436			水	PWR

22	Steam generator for a pressurized-water coolant nuclear reactor (Patent), 33:19439		水	PWR
23	Thermal-hydraulic design of LMFBR steam generators, 33:22163	米		LMFBR
24	Trends in the design of pressurized water reactor steam generators, 33:32222	チェコ	水	PWR
25	STEAM GENERATORS/FABRICATION Welding and fabrication of alloy 800 with emphasis on the development of techniques suited to steam generator plant, 33:9805	英		
26	STEAM GENERATORS/FAILURES European breeders (II): the nuclear parts are not the problem, 33:28051	米	Na	FBR
27	Evaluation of burst probability for tubes by Weibull distribusions, 33:14167 (BNL-20734)	米		
28	Failure probabilities of steam generator tubes. Annual report, 33:2484 (BNL-20368)	米		
29	Malfunctions occuring in steam and superheated water generators. Observations on the construction and equipment of these generators, 33:32519	仏		
30	Operational experience with heat exchangers in nuclear power stations with light water reactors, 33:4629		水	LWR
31	Problemof reliability of steam generators at nuclear power stations with water-moderated water-cooled reactors (PWR), 33:2298	ソ	水	PWR
32	Steam generator tube failures: world experience in water-cooled nuclear power reactors in 1974, 33:25292 (AECL-5242)	カ	水	LWR
33	Steam-generator tube failures: world experience in water-cooled nuclear power reactors in 1974, 33:32503	カ	水	LWR
34	STEAM GENERATORS/FLUID FLOW Statistical characteristics of temperature pulsations in a once-through sodium-water steam generator model, 33:32286	ソ	Na-水	
35	STEAM GENERATORS/FUNCTIONAL MODELS Dynamic model of a sodium once-through steam generator (LMFBR), 33:22166	米		LMFBR
36	STEAM GENERATORS/HEAT TRANSFER Bubble formation and heat transfer on boiling in free and forced convection, 33:29023	西独		
37	Full scale tests and thermal design correlations for coiled once-through steam generators, 33:16310 (RT/ING-(75)11)	伊		
38	Heat transfer in helical tube sodium heated steam generators, 33:28819 (CEA-CONF-3204)	仏	Na	LMFBR
39	Problems of heat transfer, pressure loss and flow stability in highly thermally stressed steam generator tubes (Nuclear power plants), 33:28192	オーストリア		
40	STEAM GENERATORS/INSPECTION Remotely controlled platform for inspection of steam generator tubes (PWR), 33:22018	米	水	PWR
41	STEAM GENERATORS/MATERIALS Carbon transfer behavior of materials for liquid-metal fast breeder reactor steam generators, 33:28045	日	Na	LMFBR
42	Comparison of various chromium--molybdenum low-alloy steels for liquid-metal fast breeder reactor steam generators, 33:28049	日	Na	LMFBR

43	Effects of service environments on the behavior of high-temperature gas-cooled reactor steam generator structural materials, 33:27996	米	He	HTGR
44	Materials performance in operating pressurized water reactor steam generators, 33:27968	米	水	PWR
	Materials in CANDU steam generators from the designer's point of view, 33:28014	カ	重水	CANDU(PHWR)
45	STEAM GENERATORS/MATHEMATICAL MODELS Dynamic simulation of an LMFBR steam generator, 33:13978 (CONF-750977-)	米	Na	LMFBR
46	Modeling of nuclear steam generator dynamics, 33:14170 (CONF-750977-)	オランダ		PWR,LMFBR MSBR
47	STEAM GENERATORS/MECHANICAL VIBRATIONS Flow-induced vibration in LMFBR steam generators: a state-of-the-art review, 33:6716 (ANL-75-16)	米		LMFBR
48	STEAM GENERATORS/ORIFICES Performance study of two-phase water tests on plug and sharp-edged type orifices (Space power reactors), 33:22292 (CNLM-5662)	米		スペースパワー
49	STEAM GENERATORS/PERFORMANCE Heat transmission and pressure drop in the steam generators heated by sodium circulation (LMFBR), 33:16286 (ANL-Trans-1024)	仏	Na	Phenix (LMFBR)
50	LMFBR steam generator performance changes at low-power operation, 33:22167	米	Na	LMFBR
51	Operating experience with Westinghouse steam generators (PWR), 33:27969	米	水	PWR
52	Tests of an experimental steam generator heated by sodium-potassium flow, 33:28195	仏	NaK	
39	STEAM GENERATORS/PERFORMANCE TESTING Full scale tests and thermal design correlations for coiled once-through steam generators, 33:11268 (RT/ING-(75)11)	伊		
37	Full scale tests and thermal design correlations for coiled once-through steam generators, 33:16310 (RT/ING-(75)11)	伊		
53	Test results prove LMFBR heat exchanger concept, 33:8731	米		LMFBR
11	STEAM GENERATORS/PICKLING The significance of the iron(III) ions for the corrosion danger in cleaning steam generators, 33:17679	西独		
39	STEAM GENERATORS/PRESSURE DROP Problems of heat transfer, pressure loss and flow stability in highly thermally stressed steam generator tubes (Nuclear power plants), 33:28192	オーストリア		
54	STEAM GENERATORS/RADIATION HAZARDS Tritium permeation through steam generator materials, 33:5280 (CONF-751026-15)	米		
55	STEAM GENERATORS/RELIABILITY Problem of reliability of the operation of steam generators of nuclear power plants with water-cooled reactors, 33:32216	ソ	水	LWR
56	STEAM GENERATORS/RESEARCH PROGRAMS Central Technical Institute (TNO): monthly report, October 1974 (LMFBR), 33:16311 (TNO-LMEC-9)	オランダ		LMFBR
57	Central Technical Institute (TNO): monthly report, November 1974 (LMFBR), 33:16312 (TNO-LMEC-10)	オランダ		LMFBR
58	Central Technical Institute (TNO): first quarterly report, 1975 (LMFBR), 33:16313 (TNO-LMEC-12)	オランダ		LMFBR

59	STEAM GENERATORS/SAFETY Tritium permeation through materials for steam generator systems, 33:20020	米		
60	STEAM GENERATORS/SPECIFICATIONS Iron oxides in the water of steam generating equipments of nuclear power plants, 33:30066	ソ		
61	STEAM GENERATORS/STEAM QUALITY Boiling crisis in a steam-generating tube heated by a liquid metal heat carrier, 33:14020	ソ	液体金属	
62	STEAM GENERATORS/STRESS CORROSION Caustic stress corrosion cracking of Inconel-600, Incoloy-800, and Type 304 stainless steel, 33:28184	英		
63	STEAM GENERATORS/SUPPORTS Reactor installation (Patent), 33:25136			
64	STEAM GENERATORS/TEST FACILITIES Development of a thermal-hydraulic test facility for full-scale LMFBR steam generator tubes, 33:22168	米	Na	LMFBR
65	Heavy components development for sodium plants at Franco Tosi, 33:32308	伊	Na	
66	STEAM GENERATORS/TESTING Steam generator test loop for sodium cooled fast breeder reactor., 33:32283	日	Na	LMFBR
67	Test report: steam generator development test program (NERVA ETS-1), 33:2381 (NTO-R-0061)	米		NERVA
68	STEAM GENERATORS/TUBES Corrosion of steam generator tubing in operating pressurized water reactors, 33:13945	米	水	PWR
69	Failure research report. Failures in welded joints of pressure parts in power plant construction, 33:9100	西独		
70	LMFBR steam generator development: duplex bayonet tube steam generator. Volume II, 33:19459 (CENC-1238(Vol.2))	米		LMFBR
71	Probe train including a flaw detector and a radiation responsive recording means with alignment means having a natural curved cast (Patent), 33:2945			
39	STEAM GENERATORS/TWO-PHASE FLOW Problems of heat transfer, pressure loss and flow stability in highly thermally stressed steam generator tubes (Nuclear power plants), 33:28192	オーストリア		
72	Two phase flow problems in power station boilers, 33:28193	英		
73	STEAM GENERATORS/ULTRASONIC TESTING Ultrasonic inspection recess in heat exchanger and nuclear steam generator tubesheets (Patent), 33:29044			
74	STEAM GENERATORS/WATER CHEMISTRY Application of disodium-hydrogen-phosphate in the passage steam generator of an English nuclear power plant, 33:32250	英		
75	STEAM GENERATORS/WATER HAMMER Analysis of water hammer observed in feedwater systems of certain PWR systems, 33:22017	米	水	PWR
76	STEAM GENERATOTS/WELDED JOINTS Feasibility studies for nondestructive testing of duplex tube-to-tubesheet joints and tubes for an LMFBR steam generator, 33:11267 (ORNL-5094)	米	Na	LMFBR

付 録 B

1967 Heat exchanger

67006
AVR
He
design,
safety

"Safety requirements fulfilled in the steam generator for the AVR nuclear power station". Johannknecht, N. (Vereinigte Kesselwer ke AG, Dusseldorf). Kerntechnik, 8:402-4 (Aug.-Sept. 1966). (In German).
NSA.21:12173

67076
MHD
vaper-gas
mass exchange

"Calculations of mass exchange in low-temperature heat exchanger for MHD installations". Styrikovich, M.A.; Godik, I.B. Izv. Akad. Nauk SSSR, Energ. Transp., No.5, 105-12 (Sept.-Oct. 1966). (In Russion).
NSA.21:9362

1968

68020
MHD
air
preheating

"Heat exchangers for air preheating in MHD conversion power plants". Borgese, Domenico; Fabre, Jean. Thermotechnica (Milan), 22:109-16 (Mar. 1968). (In Italian).
NSA.22:33592

68047
AGR
CO_2
material

"AGR materials". Hutcheon, J.M. (United Kingdom Atomic Energy Authority, Culchetch, Eng.). BNX Rev., No.2, 17-23(Nov. 1967).
NSA.22:20830

68048
AGR,HTGR
CO_2,He
material,
corrosion resi-
stance

"Some aspects of materials technology of importance to the development of nuclear reactions". Nichols, R.W. (United Kingdom Atomic Energy Authority, Culcheth(Eng.). Reactor Group); Gittus, J.H. (United Kingdom Atomic Energy Authority, Springfields (Eng.). Reactor Group). [1967]. 51p.
NSA.22:47660(CONF-671007-2)

68066
MHD
lay-out,design,
material,cost,
optimization

"Engineering problems of turbomachinery and heat transfer equipment for nuclear MHD power plants". Foerster, S.; Gilli, P.V. (Kernforschungsanlage, Juelich, Ger. Kernenergieabteilung, Vienna). Atomkernenergie, 13:205-22 (May-June 1968).
NSA.22:37503

1969

69021
HTR,AGR
He,Ne,CO_2
gas turbine
cycle, economics,
optimization

"Gas-cooled reactors with gas-turbines using helium, neon, or carbon-dioxide". Dibelius, G. (Technische Hochochule, Aachen); Foerster, S. Kerntechnik, 11: 105-10 (Feb. 1969)
NSA.23:28369

1970

70031
HTGR
He
pressure casing material

"The metallurgical properties of the Dragon reactor pressure vessel. Project Dragon". Conde, J.F.G. (Atomic Energy Establishment, Winfrith (England)). Jan 1968. 112p. NSA.24:39623 (DP-Report-80)

70026
HTGR
He
noise, coolant flow

"Noise spectra measured on the Dragon reactor primary heat exchangers". Cummins, J.D. (Atomic Energy Establishment, Winfith (England)). Sept. 1969. 41p. (DPR-632). (U.S. Sales Only). NSA.24:4083 (AEEW-R-641)

70049
HTGR
He
friction
heat transfer

"The Dragon freezer. Part II. Experimental heat transfer and friction results for single elements. Project DRAGON". Husain, L.A.; Robinson, C.H. (Atomic Energy Establishment, Winfrith (England)). Feb. 1963. 27p. NSA.24:39132 (DP-Report-126(Pt.II))

70050
HTGR
He
heat transfer
He purification

"The Dragon freezer. Part I. A theoretical investigation of the heat transfer performance. Project DRAGON". Husain, L.A. (Atomic Energy Establishment, Winfrith (England)). Jan. 1963. 26p. NSA.24:39131 (DP-Report-126(Pt.I))

70053
HTGR
He
heat transfer
safety margin
friction

"Estimate of film boiling safety margin in Dragon primary heat exchanger. Project DRAGON". Husain, L.A. (Atomic Energy Establishment, Winfrith (England)). Oct. 1962. 26p. NSA.24:39130 (DP-Report-113)

70052
HTGR
He
heat transfer, design, fabrication

"Experimental work on gaseous and boiling heat transfer associated with the design of the Dragon Primary heat exchangers and some aspects of their manufacture. Project DRAGON". Gilli, P.V. (Waagner-Biro A.G.. Vienna (Austria). Atomic Energy Establishment, Winfrith (England)). Oct 1963. 60p. (CONF-631012-7). NSA.24:52953 (DP-Report-222)

70055
HTGR
He
corrosion and fretting problem

"O.E.C.D. high temperature reactor project (DRAGON). Tenth Annual Report, 1968-69". (Organization for Economic Co-operation and Development, Paris (France)). 175p. (U.S. Sales Only) NSA.24:18674 (NP-18148)

70041
THTR
He, Air
testing

"Final report on the tests on a helically wound heat exchanger for the steam generator of the THTR". Scholz, F.; Meis, Th.; Grochn. H.G. (Kernforschungsanlage, Juelich (West Germany). Institut fuer Reaktorbauelemente). Mar. 1970. 78p. (In German and English). (U.S. Sales Only). NSA.24:36455 (JUL-649-RB)

1971

71046
HTR
He
heat transfer,
high-pressure

"Preliminary experiments of a heat exchanger in helium under high-pressure at the KFA Juelich". Groehn, H.G.; Scholz, F.; Juerk, G. (Kernforschungsanlage, Julich, Ger.). Atomrvirt., Atomtech.; 15:576-7 (Dec 1970). (In German). NSA.25:20780

1972

72003
HTGR
He
corrosion,
tube failure

"Corrosion experience with the Dragon primary heat exchangers". Gray, P.S; Salter, W.B. (Dragon Operations Branch, Winfrith, Eng.). J. Brit. Nucl. Energy Soc.; 10:No.3, 207-15 (Jul 1971). NSA.26:449

72028
MHD
material

"Heat exchange processes in exchanger materials. High temperature heat transfer equipment". Yerouchalmi, David (Lille Univ., Villeneuved'Ascq. (France)). 1972. 187p. (In French). (U.S. Sales Only). NSA.26:56023 (FRNC-TH-231)

72055
AGR,THR
cross-flow,
heat transfer
surfaces

"Development of improved heat transfer surfaces for tubes in cross-flow, using a large scale model technique". Neal, S.H.C.; Hitchcock, J.A. (Central Electricity Research Labs., Leatherhead, Eng.). pp FC7.8.1-11 of Heat Transfer 1970. Vol. III./Grigull, Ulrich (ed.). Amsterdam; Elsevier Publishing Co. (1970). NSA.26:25469

1973

73004
HTGR
He
anti-fretting
design, tube
vibration

"Fretting in tube supports of heat exchangers". Fricker, H.W. pp213-33 of Component Design in High Temperature Reactors Using Helium as a Coolant. London ; Institution of Mechanical Engineers (1972). NSA.29:14042

73024
HTGR
He
flow
instability

"Theoretical analysis of flow instabilities in the Dragon primary heat exchangers. Dragon Project". Davies, A.L. (Atomic Energy Establishment, Winfrith (England)). Jun 1969. 28p. (U.S. Sales Only) NSA.27:29468 (DP-Report-651)

73038
HTGR
anti-fretting
design, tube
vibration

"Fretting in tube supports of heat exchangers". Fricker, H.W. (Winterthur Engineering Works Div., Switzerland). Sulzer Tech. Rev.; No. Nuclear 72, 55-62 (1972). NSA.27:22412

73047
HTR
He
cost

"Cost analysis of components for nuclear power plants with high-temperature reactor and gas-turbine". Schlenker, H.V. (Kernforschungsanlage, Juelich (West Germany). Institut fuer Reaktorentwicklung). Feb 1973. 157p. (In German). (U.S. Sales Only). NSA.28:7296(JUL-920-RG)

73045
HTR
contamination, circuit activity

"Calculation of the circuits activity in a high temperature reactor". Brisbois, J.; Beslu, P.; Beslu, P.; L'homme, A. (Commissariat a l'Energie Atomique, Saclary (France). Centre d'Etudes Nucleaires). [nd]. 31p. (In French). (U.S. Sales Only). NSA.28:7285 (CEA-CONF-2214)

73052
MHD
regenerative heat exchanger

"High temperature regenerative heat exchangers for open cycle MHD power plant. II". Imatake, T.; Maita. Y.; Matsubara, S.; Irino, M. (Mitsubishi Heavy Industries, Ltd., Kobe, Japan). Mitsubishi Juko Giho; 10:No.3, 400-408 (May 1973). (In Japanese) NSA.28:27160

73065
MHD
regenerative heat exchanger

"High temperature regenerative heat exchangers for open cycle MHD power plant. I.". Imatake, T.; Matsubara, S.; Irino, M.; Maita, Y. (Mitsubishi Heavy Industries, Ltd., Tokyo). Mitsubishi Juko Giho ; 10:No.1, 115-131 (Jan 1973). (In Japanese). NSA.28:18102

1974

74004
HTGR
ceramic heat exchanger

"Use of nonmetallic materials with high-temperature gas-cooled reactors with special reference to heat exchangers for power and process plants". Hryniszak, W. (Clarke Chapman-John Thompson Ltd., Gateshead (UK)). 6 Jun 1973. 47p. (U.S. Sales Only). NSA.29:11790 (CONF-730668-1)

74007
HTGR
He
gas turbine, configuration

"HTGR gas turbine power plant configuration studies". Adams, R.G.; Bell, F.R.; McDonald, C.F.; Morse, D.C. (Gulf General Atomic Co., San Diego, Calif. (USA)). 26 Jul 1973. Contract AT(04-3)-167. 37p. (CONF-731105-10). NSA.29:31193(GULF-GA-A-12651)

74014
He
design, loop

"High temperature helium heat exchanger loop". (Ishikawajima-Harima Heavy Industries Co. Ltd., Tokyo). Ishikawajima-Harima Giho; 13:No.3, 349-357 (May 1973). (In Japanese). NSA.29:14690

74067
HTGR
He
design, material

"Heat exchangers for high-temperature gas cooled reactors". Okamoto, Y.; Shimomura, H.; Hishida, M. (Japan Atomic Energy Research Inst., Tokai). Kinzoku Zairyo; 13:No.5, 75-82 (May 1973). (In Japanese). NSA.30:1881

74073
HTGR
He
heat resisting alloy

"Present situation of refractory metals and alloys for use in nuclear energy. II. Present Situation and Future Prospect of Refractory Metals and Alloys". Maeda, T. (Kawasaki Heavy Industries, Ltd., Kobe, Japan). FAPIG (Tokyo); No.67, 166-176; 204 (Nov 1972). NSA.30:7285

1975

75014
HTGR
He
heat transfer, static stability

"Core anuxiliary heat exchanger for a high-temperature gas-cooled reactor; water side static stability analysis and design". Carosella, D.P. Jr.; Harvego, E.A. 74-WA/HT-7. New York; American Society of Mechanical Engineers (1974). 8p. NSA.31:31777

75016
HTGR
He
heat transfer, radiation effect

"Radiation effects on heat transfer in reactor core and heat exchangers of HTGR". Mori, Y.; Hijikata, K.; Yamada, Y. 74-WA/HT-2. New York; American Society of Mechanical Engineers (1974). 13p. NSA.31:31775

75021
HTGR
He
performance, stress analysis

"HTGR safety studies progress report for the period ending March 31, 1974". Sanders, J.P. (Oak Ridge National Lab., Tenn. (USA)). Oct 1974. Contract W-7405-eng-26. 100p. (GCR-DL-74-1). NSA.31:2516(ORNL-TM-4736)

75022
HTGR
He
performance

"HTGR safety studies quarterly progress report for the period ending June 30, 1974". Sanders, J.P. (Oak Ridge National Lab., Tenn. (USA)). Oct 1974. Contract W-7405-eng-26. 50p. (GCR-DL-74-2). NSA.31:2517 (ORNL-TM-4737)

75032
HTR
He
material, hydrogen permeation

"Investigations on HTR-heat exchanger materials". Serpekian, T. (Kernforschungsanlage Juelich G.m.b.H. (F.R. Germany). Inst. fuer Reaktorentiwicklung). Sep 1974. 174p. (In German). (U.S. Sales Only). NSA.32:2416 (JUL-1111-RG)

75043
HTGR
He
heat transfer

"Analytical treatment of two-dimensional heat conductivity in a rectangular cross section under given boundary conditions (HTGR heat exchangers)". Achenbach, E.; Iniotakis, N. (Kernforschungsanlage Juelich G.m.b.H. (F.R. Germany). Inst. fuer Reaktorbauelemente). Mar 1975. 14p. (In German). (U.S. Sales Only). NSA.32:10756 (JUL-1172)

75045
HTGR
He
heat transfer hydrogen permeation, creep

"Preliminary study on high temperature heat exchanger for nuclear steel making". Ikegami, H. (Nippon Kokan K.K., Tokyo); Mori, Y. Testu To Hagane; 60:No.8, 1166-1178 (Jul 1974). (In Japanese). NSA.32:18841

1976

76026
HTGR
He
heat transfer
radiation effect

"Radiation effects on heat transfer in the reactor core and heat exchangers of an HTGR". Mori, Y.: Hijikata, K.; Yamada, Y. (Tokyo Inst. of Tech.). J. Heat Transfer; 97:No.3, 400-405 (Aug 1975). NSA.33:6707

76036
HTGR
He
performance

"Thermal transport properties of helium, helium-air mixtures, water, and tubing steel used in the CACHE program to compute HTGR auxiliary heat exchanger performance". Tallackson, J.R. (Oak Ridge National Lab., Tenn. (USA)). Feb 1976. Contract W-7405-eng-26. 110p.
NSA.33:25144(ORNL/TM-4931)

Steam generator

1967

67003
HTGR
He
design,
steam/water
flow stability

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending September 30, 1965". (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Dec. 31, 1965. Contract AT(04-3)-633. 156p. NSA.21:38447 (GA-6830)

67022
HTGR
design

"High temperature gas cooled reactors with steam cycles". Stadie, K.B. (European Nuclear Energy Agency, Paris (France)). 20p. NSA.21:34086 (EUR-2757.e,pp21-)

67036
HTGR
He
development,
design,
testing

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending March 31, 1966". (General Dynamics Corp., San Diego, Calif. General Atomic Div.). May 11, 1966. Contract AT(04-3)-633. 116p. NSA.21:27670 (GA-7086)

67051
HTGR
He
flow
distribution

"Flow model tests of the reactor outlet plenum for the 330 MW(e) HTGR". Hausermann, R.; Malek, G.J.; Meyer, L. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Feb. 24, 1967. Contract AT (04-3)-633. 145p. NSA.21:27671 (GA-7638)

67067
HTGR
He
performance

"40-MW(e) prototype high-temperature gas-cooled reactor postconstruction research and development program. Quarterly Progress Report for the Period Ending October 31, 1966". (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Feb. 2, 1967. Contract AT(04-3)-314. 38p. NSA.21:25624 (GA-7595)

67070
HTGR
power
density

"Power density of integral steam generators for high-temperature gas-cooled reactors". Gilli, P.V. (Waagner-Biro AG, Vienna). Neue Tech. 9: 10-17 (Feb. 1967). NSA.21:22094

67074
HTGR
He
flow stability
tube vibration

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending June 30, 1966". (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Sept. 30, 1966. Contract AT(04-3)-633. 134p. NSA.21:38451 (GA-7314)

67075
HTGR
He
flow stability,
material,
tube vibration

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending Dec. 31, 1965". (General Dynamic Corp., San Diego, Calif. General Atomic Div.). Mar. 21, 1966. Contract AT(04-3)-633 109p. NSA.21:38449 (GA-6950)

1968

68002
AGR
CO_2
configuration

"Hinkley Point B : Turbines and generators". Harris, F.R.; Creek, F.R.L. (General Electric Co., Ltd., Wembley, Eng.). Nucl. Eng.. 13:661-3 (Aug. 1968). NSA.22:46970

68005
AGR
CO_2
construction

"Dungeness B : Membrane moves in". Engineering, 205: 440-2 (Mar. 22, 1968) NSA.22:22728

68011
AGR
CO_2
description

"Gas-cooled reactors with a novel boiler concept". Churchill, G.F. Neue Tech., 9:205-8 (Oct. 1967). NSA.22:3852

68020
AGR
CO_2
design,
material

"Steam raising unit design". Warren, L.W. (International Combustion Ltd., Derby, Eng.). Proc. Inst. Mech. Eng. (London), 181:Pt. 31, 217-26 (1966-67). NSA.22:7857

68056
AGR
CO_2
fabrication
weld

"Boiler element fabrication for the advanced gas-cooled reactor at Dungeness "B" nuclear power station". Gifford, A.M.; Thrower, A. Weld. Metal Fabr., 36:327-32 (Sept. 1968). NSA.22:50767

68099
AGR
CO_2
vibration,
corrosion,
support system

"Dungeness B: The boiler conflict". Engineering, 206: 392-3 (Sept. 1968). NSA.22:53297

68058
HTGR
He
stress corrosion

"Peach Bottom stress corrosion cracking". Hull, L.J.; Watson, J.F. (Gulf General Atomic Inc., San Diego, Calif.). Trans. Amer. Nucl. Soc., 11: NSA.22:32840

68076
HTGR
He
performance
testing

"Engineering tests during the initial operation of the Peach Bottom HTGR". Kantor, M.E.; Menzel, H.F.; Schlicht, R.W. (Gulf General Atomic Inc., San Diego, Calif.). Contract AT(04-3)-314 Nucl. Eng. Design, 7:297-311 (1968). NSA.22:32809

68079
HTGR
He
performance
testing

"Engineering tests during the initial operation of the Peach Bottom HTGR". Kantor, M.E.; Menzel, H.F.; Schricht, R.W.; Westlake, W.J. Jr. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). June 16, 1967. 29p. NSA.22:30148 (GA-8040)

68080
HTGR
He
performance

"40-MW(e) prototype high-temperature gas-cooled reactor postconstruction research and development program. Quarterly Progress Report for the Period Ending July 31, 1967". (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Aug. 30, 1967. 60p.
NSA.22:12023 (GA-8153)

6811
MHD
lay-out,
design,
material,
cost,
optimization

"Engineering problems of turbomachinery and heat transfer equipment for nuclear MHD power plants". Foerster, S.; Gilli, P.V. (Kernforschungsanlage, Juelich, Ger. Kernenergieabteilung, Vienna). Atomkernenergie, 13: 205-22 (May-June 1968). NSA.22:37503

1969

69005
HTGR
He
velocity
profile,
temperature
contour

"PSC region I flow model (series 2) test report". Walker, W.E. (Gulf General Atomic, Inc., San Diego, Calif.). July 8, 1968. 90p.
NSA.23:4144 (GAMD-8625)

69010
THTR
He
design
construction

"Design and construction of the THTR-prototype nuclear power plant". Hennings, U. (BCC/Krupp, Mannheim, Ger.) : Schmiedel, F. Atomwirt., Atomtech., 14:118-21 Mar. 1969). (In German). NSA.23:21439

69031
THTR
He
design

"Possible solutions for integrated steam generators of the THTR". Hennings, U. (Brown Boveri/Krupp, Mannheim, Ger.). pp EN.1/161-22 of Symposium on the Technology of Integrated Primary Circuits for Power Reactors, Proceedings, Paris, 20th-22nd May 1968. Paris, European Nuclear Energy Agency, 1968.
NSA.23:28479

69025
HTGR
He
design

"Boilers in the high-temperature gas-cooled reactor". Cameron, P.J.: O'Tallamhain, C. (Nuclear Power Group Ltd., Knutsford, Eng.). pp 571-83 of Advanced and High-Temperature Gas-Cooled Reactors. Vienna, International Atomic Energy Agency, 1969.
NSA.23:37606

69043
HTGR
design
fabrication

"Possibilities of the industrial construction of the principal components of a high-temperature power reactor". Brown, B.D.; Audinet, M. Translated for Oak Ridge National Lab., Tenn., from IAEA Symposium on Advanced and High Temperature Gas-Cooled Reactors, Juelich, Germany, 21-25 October 1968. 25p.
NSA.23:15603 (ORNL-tr-3010)

69050
HTGR
He
flow stability

"Public Service Company of Colorado 330-ME(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending June 30, 1969". (Gulf General Atomic, Inc., San Diego, Calif.). July 31, 1969. Contract AT(04-3)-633 233p. NSA.23:47169 (GA-9440)

69074
HTGR
He
design

"HTGR base program quarterly progress report for the period ending February 28, 1967". (Gulf General Atomic, Inc., San Diego, Calif.). Apr. 20, 1967. Contract AT(04-3)-167. 177p.
NSA.23:4166 (GA-7801)

69083
HTGR
He
testing,
vibration

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending September 30, 1966". (Gulf General Atomic, Inc., San Diego, Calif.). Feb. 7, 1967. Contract AT(04-3)-633. 176p.
NSA.23:9117 (GA-7453)

69084
HTGR
He
testing,
vibration

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending December 31, 1966". (Gulf General Atomic, Inc., San Diego, Calif.). Mar. 20, 1967. Contract AT(04-3)-633. 154p.
NSA.23:9118 (GA-7634)

69098
HTGR
He
weldments

"Gass-cooled reactor program". Coobs, J.H. (Oak Ridge National Lab., Tenn.). (ORNL-4370, pp 165-74).
NSA.23:8276 (ORNL-4370)

69017
AGR
CO_2
design,
engineering
and
economic
factors

"Boilers for advanced gas cooled reactors". Merrett, D.J. (Atomic Power Constructions Ltd., Sutton, Eng.); Shaw, T. pp EN.1/32.1-17 of Symposium on the Technology of Integrated Primary Ciruits for Power Reactors, Proceedings, Paris, 20th-22nd May 1968. Paris, European Nuclear Energy Agency, 1968.
NSA.23:28481

69024
AGR
CO_2
design

"The pod boiler AGR circuit". Seymour, D.F. (Nuclear Design and Construction Ltd., Whetstone, Eng.). pp EN.1/34.1-17 of Symposium on the Technology of Integrated Primary Circuits for Power Reactors, Proceedings, Paris, 20th-22nd May 1968. Paris, European Nuclear Energy Agency, 1968.
NSA.23:28482

1970

70009
HTGR
He
design criteria

"A review of design criteria for the steam generator of the 330-MW(e) high temperature gas-cooled reactor plant of the Public Service Company of Colorado". Schuetzenduebel, W.G.; Hunt, P.S. (Gulf General Atomic, Inc., San Diego, Calif.). 15 May 1969. Contract AT(04-3)-633. 42p. (CONF-69114-1).
NSA.24:9549 (GA-9342)

70015
HTGR
He
design

"High-temperature gas-cooled reactor: steam generator and circulator". Yampolsky, J.S. (Gulf General Atomic Inc., San Diego, Calif.). Proc. Amer. Power Conf.; 31:258-67(1969). NSA.24:18643

70027
HTGR
He
design

"Fort Saint Vrain nuclear power station". Walker, R.E.; Johnson, T.A. Nucl. Eng. Int.; 14:1069-77 (Dec 1969). NSA.24:9063

70052
HTGR
He
flow st-bility

"Public Service Company of Colorado 330-MW(e) high-temperature gas-cooled reactor research and development program. Quarterly Progress Report for the Period Ending September 30, 1968". (Gulf General Atomic. Inc., San Diego, Calif.). 31 Oct 1968. Contract AT(04-3)-633. 194p.
NSA.24:28985 (GA-8879)

70095
HTGR
He
tube support

"Tube support damping Fort St. Vrain steam generator". Kellar, W. (Gulf General Atomic, Inc., San Diego, Calif.). 15 Nov 1969. Contract AT(04-3)-633. 26p.
NSA.24:11095 (GAMD-9735)

70104
HTGR
He
weld, corrosion resistance

"Fuel element development". (Oak Ridge National Lab., Tenn.). (ORNL-4508, pp 3-70).
NSA.24:24835 (ORNL-4508)

70074
HTGR
He
performance

"Operating experience with the Dragon steam generator". Holzner, J. (Waagner-Biro AG, Vienna); Gilli, P.V.; Gray, P.S.; Mclver, R.F.; Green, I.S. pp 301-19 of Performance of Nuclear Power Reactor Components. Vienna; IAEA (1970). NSA.24:28875

70105
HTGR
He
weld, corrosion resistance

"High-temperature gas-cooled reactor development". (Oak Ridge National Lab., Tenn.). (ORNL-4424, pp 3-72).
NSA.24:4060 (ORNL-4424)

70024
HTR
He
design

"The civil H.T.R. reference design study. Project DRAGON". Thorn, J.D. (ed.). (United Kingdom Atomic Energy Authority, Risley (England). Reactor Group). Jan 1963. 204p. NSA.24:38490 (DP-Report-135(Dt.2))

70028
AGR
CO_2
design

"Advanced nuclear plant design". Drakes, D.F. (British Nuclear Design and Construction Ltd., Whetstone, Eng.). Elec. Eng. (Melbourne); 46:30-1 (Jun 1969). NSA.24:28882

70029
AGR
CO_2

"Hartlepool AGR : Pod boilers". George, B.V. (British Nuclear Design and Construction, London). Nucl. Eng. Int.; 14:983-4 (Nov 1969). NSA.24:9054

1971

71022
HTGR
He
design

"Design detailes of the boiler for a 330MW HTGR power station". Schutzendubel, W.G. Energie; 22";;3-20 (Apr 1970). (In German). NSA.25:28631

71025
HTGR
He
design
fabrication

"Steam generators of Fort St. Vrain nuclear power plant, USA". Kaelin, W. Sulzer Tech. Rev.; 52:167-74 (1970). NSA.25:25956

71036
HTGR
He
development

"Steam generators for high-temperature gas-cooled reactor plants in the U.S.A.". Schuetzenduebel, W.G. (Gulf General Atomic, Inc., San Diego, Calif.). 19 Mar 1971. Contract AT(04-3)-633. 135p. NSA.25:31623 (GA-10338)

71092
HTGR
He
weld,corrosion resistance

"Gas-cooled reactor program semiannual progress report for period ending March 31, 1970". (Oak Ridge National Lab., Tenn.). Nov 1970. Contract W-7405-eng-26. 158p. NSA.25:5990 (ORNL-4589)

71019
THTR
He
design

"Design of the steam generators for the 300 MW THTR project". (European Atomic Energy Community. Brown Boveri/Krupp Reaktorbau GmbH, Mannheim (West Germany). Kernforschungsanlage, Juelich (West Germany)). 1969. 26p. (In German). (U.S. Sales Only). NSA.25:46287 (EUR-4336)

71023
HTGR
design,
optimum,
stability
characteristic

"Design of steam generators for a high temperature nuclear power plant". Hansen, Ulf (Kernforschungsanlage, Juelich, Ger.). Brennst.-Waerme-Kraft; 22:421-6 (Sep 1970). (In German). NSA.25:1916

71087
HTGR
He
material,
tube

"Materials for the boiler tubes of a HTGC power reactor". Project DRAGON. Conde, J.F.G. (Atomic Energy Establishment, Winfrith (England)). May 1967. 54p.
NSA.25:4748 (DP-Report-537)

1972

72006
HTGR
helical steam
generator,
characteristics

"Helix steam genrators for gas-cooled reactor system". Luescher, O. (Sulzer Brothers Ltd., Winterthur, Switzerland). Nucl. Eng. Int.; 17:No.194,543-6 (Jul 1972). NSA.26:58319

72015
HTGR
He
deposit

"Chemistry of HTGR". de Nordwall, H.J. (Oak Ridge National Lab., Tenn.). 1972. 44p.
NSA.26:45288 (TID-20106)

72088
HTGR
He
tube lifetime

"Effect of an oscilating dryout point on evaporator tube lifetime". Lasarev, M. (Gulf General Atomic, San Diego, Calif.). Trans. Amer. Nucl. Soc.; 15: No.1, 396-7 (Jun 1972). NSA.26:44752

72093
HTGR
He
flow-induced
vibration

"Public Service of Colorado steam generator air flow test". Kaufman, J.S. (Gulf Energy and Environmental System, San Diego, Calif.). Trans. Amer. Nucl. Soc.; 14:No.2, 786-7 (Oct 1971). NSA.26:2136

72098
HTR
CO_2
deposit,
oxidation,
hydrogen
permeation

"Investigation of the carbon deposition on HTR steam generator materials and of the diffusion of hydrogen through such materials". Serpekian, T. (Kernforschungsanlage, Juelich, Ger.). pp 403-6 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforun E.V. (1971). (In German).
NSA.26:42519

72029
HTR
He
design

"Divided steam system for HTR". Nucl. Eng. Int.; 16:No.184, 788-9 (Sep 1971). NSA.26:6320

1973

73009
HTGR
He
design,
material

"Boilers for H.T.R. application". Whitehouse, P.R. pp 163-76 of Component Design in High Temperature R Reactors Using Helium as a Coolant. London : Institution of Mechanical Engineers (1972). NSA.27:14040

73052
HTGR
He
testing,
heat transfer,
helices design

"Investigations on steam generator models of in-line tube arrangement and multistart helices design in pressurized air and helium". Groehn, H.G.; Scholz, F. pp 193-212 of Component Design in High Temperature Reactors Using Helium as a Coolant. London; Institution of Mechanical Engineers (1972). NSA.27:14041

73020
AGR
CO_2
development,
material,
design

"Development problems of steam generators in the context of the future development of nuclear power plants". Hryniszak, W. (Clarke, Chapman and Co. Ltd., Gateshead, Eng.): Bitterlich, E. Mitt. Ver. Grosskesselbetr.; 51:No.6, 444-54 (1971). (In German). NSA.27:4420

73024
AGR
CO_2
feed pump
design

"Feed pump design concept for 660 MW generating sets". Richardson, J. (Weir Pumps Ltd., Alloa. Eng.); Taylor, J.M. Proc. Inst. Mech. Eng. (London); 184:No.3 N, 1-10 (1969). NSA.27:21647

73078
THTR
performance
testing,
testing cost

"Test requirements and testing of pressure systems for nuclear power stations. Experience with the THTR Linear and Steam Generators". Schleimer, W. Atomwirt., Atomtech.; 18:No.6, 280-285 (1973). (In German). NSA.28:20423

1974

74013
HTGR
He
design

"Large HTGR steam generator design verification and support programs Summary". Schultz, G.M. (Gulf General Atomic Co., San Diego, Calif. (USA)). 15 Dec 1973. 44p. NSA.29:28639 (GULF-GA-A-12808 (Ud.1))

74034
HTGR
He
vibration

"Large HTGR steam generator design verification and support programs Vibrations". Kaufman, J.S. (Gulf General Atomic Co., San Diego, Calif. (USA)). 15 Dec. 1973 61p. NSA.29:28641 (GULF-GA-A-12808 (Vol.4))

74035
HTGR
He
vibration

"PSC steam generator hot flow test". Kaufman, J.S. (Gulf General Atomic Co., San Diego. Calif. (USA)). 19 Sep 1973. Contract AT(04-3)-633 19p. NSA.29:14611 (GULF-GA-A-12764)

74052
HTGR
He
heat transfer,
flow, stability

"Large HTGR steam generator design verification and support programs Heat Transfer, Fluid Flow, and Stability". Schultz, G.M. (Gulf General Atomic Co., San Diego, Calif. (USA)). 15 Dec 1973 155p. NSA.29:28640 (GULF-GA-A-12808(Vol.2))

74100
HTR,AGR
transient
behavior,
model

"Modeling of once-through boiler dynamics". Taylor, E. (Nuclear Power Group Ltd., Risley, Eng.). pp Paper No.10 of Boiler Dynamics and Control in Nuclear Power Stations, Proceedings of the International Conference Held on 22-23 March 1973 at the Institution of Civil Engineers, London, Eng,; British Nuclear Energy Society (1973).
NSA.30:25848

74074
AGR
CO_2
transient
behavior

"Operational and fault transient studies of a once-through AGR boiler using the OSCA digital computer program". Bickers, B.H.; Carstairs, R.L. (Atomic Power Construction Ltd., Sutton, Eng.). pp Paper No. 11 of Boiler Dynamics and Control in Nuclear Power Stations, Proceedings of the International Conference Held on 22-23 March 1973 at the Institution of Civil Engineers, London. London, Eng.; British Nuclear Energy Society (1973). NSA.30:25849

74135
AGR
CO_2
flow stability

"Study of oscillatory instabilities in the parallel channels of a high pressure once-through boiler rig". Fallows, T.; Hitchcock, J.A.; Jones, R.C.; Lis, J.; Northover, E.W. (Central Electricity Research Labs., Leatherhead, Eng.). pp Paper No.14 of Boiler Dynamics and Control in Nuclear Power Stations, Proceedings of the International Conference Held on 22-23 March 1973 at the Institution of Civil Engineers, London. London. Eng.; British Nuclear Energy Society (1973).
NSA.30:25850

74134
HTR
flow stability

"Flow stability and dynamic behavior of nuclear". Beckmann, G. (Waagner-Biro A.G., Vienna); Fritz, K.; Lendl, H.; Gilli, P.V. pp Paper No.15 of Boiler Dynamics and Control in Nuclear Power Stations, Proceedings of the International Conference Held on 22-23 March 1973 at the Institution of Civil Engineers, London. London, Eng.; British Nuclear Energy Society (1973). NSA.30:25851

74112
HTGR
He
material

"Large HTGR steam generator design verification and support programs: Materials". Mazandarany, F.N.; Rob Roberts, D.I. (General Atomic Co., San Diego, Calif. (USA)). 15 Dec. 1973. 53p.
NSA.30:25839 (GULF-GA-A-12808 (Vol.3)).

1975

75003
HTGR
He
fabrication,
inspection

"Steam-generator installation equipment". (Aerojet-General Corp., Sacramento, Calif. (USA). Nuclear Div.). Nov 1968. 15p. NSA.31:24632 (TID/SNA-177)

75005
HTR
tube material

"Stainless steel and nickel alloy tubes for nuclear power generation". Egnell, L. (Sandvik AB, Sweden). J. Brit. Nucl. Energy Soc.; 13:No.4, 375-81 (Oct 1974). NSA.31:17117

75039
HTGR
He
vibration

"Large HTGR steam generator design verification and support programs Vibration". Blevins, R.D.; Burton, T.E.; Cheng, S.C.; Kaufman, J.S. (General Atomic Co., San Diego, Calif. (USA)). 30 Sep 1974. 86p. (GA-LTR-5(Amend.1)). NSA.31:5190 (GA-A-12808 (Vol.4))

75053
HTGR
He
inspection, maintenance

"Volume I. System Analysis. Steam Generator Plenum Service System. Phase I. Preliminary Design. Final Summary Report to Gulf General Atomic, Inc.". (Aerojet-General Corp., Sacramento, Calif. (USA). Nuclear Div.). [nd]. 273p.
NSA.31:24625 (NDS-5(Vol.1))

75054
HTGR
He
remote handling

"Volume V. Equipment Operation and Engineering Analysis. Steam Generator Plenum Service System. Phase 1. Preliminary Design. Final Summary Report to Gulf General Atomic, Inc". (Aerojet-General Corp., Sacramento, Calif. (USA). Nuclear Div.). [nd]. 183p. (Phase 1)). NSA.31:24629 (NDS-5(Vol.4) (Phase1))

75055
HTGR
He
remote handling equipment

"Volume II. Equipment Design Criteria and Descriptions. Steam Generator Plenum Service System. Phase 1. Preliminary Design. Final Summary Report to Gulf General Atomic, Inc.". (Aerojet-General Corp., Sacramento, Calif. (USA). Nuclear Div.). [nd]. 280p.
NSA.31:24626 (NDS-5(Vol.2) (Ph.1))

75056
HTGR
He
remote handling equipment

"Volume VI. Program Planning, Phase II Through V. Steam Generator Plenum Service Sysrem. Phase 1. Preliminary Design. Final Summary Report to Gulf General Atomic, Inc". (Aerojet-General Corp., Sacramento, Calif. (USA). Nuclear Div.). [nd].
NSA.31:24630 (NDS-5(Vol.6))

75057
HTGR
He
remote viewing equipment

"Proposal No.681023 to Gulf General Atomic Inc. for steam steam generator plenum viewing system". (Aerojet-General Corp., Sacramento, Calif. (USA). Nuclear Div.). Sep 1968. 96p.
NSA.31:24631 (TID/SNA-174)

75084
HTGR
He
corrosion, material

"Effects of service environments on the behavior of HTGR steam generator structural materials". Mazandarany, F.N.; Rittenhouse, P.L. (General Atomic Co., San Diego, Calif. (USA)). 15 Jul 1975. Contract AT(04-3)-167. 41p. (CONF-750911-6).
NSA.32:25250 (GA-A-13553)

75093
HTGR
design

"Clean power : for the nuclear future". Nelken, H.H.; Barratt, R.O.; Mackenzie, S.J.; Essebaggers, J.; Gardner, W.A.; Walker, K.L. (Foster Wheeler Corp., Livingston, NJ). Heat Eng. (Livingston, N.J.); 45:No.8, 129-144 (1972). NSA.32:23992

75103
HTGR
He
heat transfer, flow, stability

"Large HTGR steam generator design verification and support programs". Duncan, D.S. (General Atomic Co., San Diego, Calif. (USA)). 10 Mar. 1975. 44 p. (GA-LTR-5). General Electric, San Diego, CA. NSA.32:5355 (GA-A-12808)

75105
HTGR
He
stability

"Stability behavior of steam generatots with upward flow of helium. Final Report. Dragon Project Report". Fritz, K.; Gilli, P.V. (UKAEA Reactor Group, Winfrith. Atomic Energy Establishment). Apr 1969. 33p. NSA.32:27237 (DP-Report-625)

1976

76007
HTGR
He
corrosion, material

"Behavior of metals in high-temperature reactor helium for steam generator". Bates, H.G.A.; Betteridge, W.; Cook, R.H.; Graham, L.W.; Lupton, D.F. (United Kingdom Atomic Energy Establishment, Winfrith, Eng.). Nucl. Technol.; 28:No.3, 424-440 (Mar 1976). NSA.33:27997

76017
HTGR
He
design

"High-temperature gas-cooled reactor steam generator desig design". Schuetzendeulbel, W.G. (General Atomic Co., Sa San Diego, CA). Nucl. Technol.; 28:No.3, 315-327 (Mar 1976). NSA.33:27999

76043
HTGR
He
corrosion, material

"Effects of service environments on the behavior of high-temperature gas-cooled reactor steam generator structural materials". Mazandarany, F.N. (General Atomic Co., San Diego, CA); Rittenhouse, P.L. Nucl. Technol,; 28:No.3, 406-23 (Mar 1976). NSA.33:27996