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Nuclear Reactors and Technology

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PURPOSE

Nuclear Reactors and Technology (NRT) announces on a monthly basis the current worldwide information available from the open literature on nuclear reactors and technology, including all aspects of power reactors, components and accessories, fuel elements, control systems, and materials.

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- Auxiliary, Mobile, Package, and Transportable

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Nuclear Reactors and Technology

POWER REACTORS

Light-Water Moderated, Boiling Water Cooled

1
(CONF-9010224-1)
Application of Galerkin's method for calculating boiling water reactor limit-cycle amplitude using the LAPUR feedback-transfer function and the point-kinetics equations. Damiano, B , March-Leuba, J A , Euler, J A (Oak Ridge National Lab , TN (USA)) [1990] Contract AC05-84OR21400 13p NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91001517
From International workshop on BWR stability, Upton, NY (USA) (18 Oct 1990)
This paper describes a technique for calculating boiling water reactor (BWR) behavior during steady-state limit-cycle oscillations An approximate solution is obtained from the application of Galerkin's method to a BWR dynamic model consisting of the point-kinetics equations and the LAPUR calculated power to-reactivity feedback-transfer function The approximate-solution technique is described, and comparisons of approximate solutions with numerical results and measured data are given 7 refs , 5 figs

2
(CONF-9010224-2)
Radial nodalization effects on BWR [boiling water reactor] stability calculations. March-Leuba, J (Oak Ridge National Lab , TN (USA)) [1990] Contract AC05-84OR21400 10p NTIS, PC A02/MF A01, OSTI, INIS, GPO Dep Order Number DE91002662
From International workshop on BWR stability, Upton, NY (USA) (18 Oct 1990)
Computer simulations have shown that stability calculations in boiling water reactors (BWRs) are very sensitive to a number of input parameters and modeling assumptions In particular, the number of thermohydraulic regions (i e , channels) used in the calculation can affect the results of decay ratio calculations by as much as 30% This paper presents the background theory behind the observed effects of radial nodalization in BWR stability calculations The theory of how a radial power distribution can be simulated in time or frequency domain codes by using "representative' regions is developed The approximations involved in this method of solution are reviewed, and some examples of the effect of radial nodalization are presented based on LAPUR code solutions 2 refs , 4 figs , 2 tabs

3
(EPRI-NP-6998-M)
Laboratory examination of tubes R35C70 and R36C67 removed from the V.C. Summer Nuclear Station. Frye, C R (Electric Power Research Inst , Palo Alto, CA (USA), Babcock and Wilcox Co , Lynchburg, VA (USA) Research and Development Div) Oct 1990 45p Research Reports Center, Box 50490, Palo Alto, CA 94303
Two tubes from V C Summer Nuclear Station were examined The examination revealed extensive primary water stress corrosion cracking (PWSCC) which occurred at the over laps between the discrete rolling steps used for expanding the tube In addition to the PWSCC on the ID surfaces of the tube OD initiated IGA and stress corrosion cracking was discovered in the crevice regions where the tubes passed through the flow distribution baffle and first tube support plate The data collected did not allow a determination of the events that led to initiation of the OD defects in this region The eddy current techniques used to size the defects in the field and the lab are also evaluated 9 refs , 4 figs , 7 tabs

4
(EPRI-NP-7026-D)
Assessment of typical BWR [boiling water reactor] vessel configurations and examination coverage. Walker, S M , Feige, E J , Ingamells, J R , Calhoun, G L , Davis, J , Kapoor, A (Electric Power Research Inst , Palo Alto, CA (USA), Jones (J A) Applied Research Co , Charlotte, NC (USA), EPRI Nondestructive Evaluation Center, Charlotte, NC (USA), Southwest Research Inst , San Antonio, TX (USA), Westinghouse Electric Corp , Pittsburgh, PA (USA)) Oct 1990 105p Research Reports Center, Box 50490, Palo Alto, CA 94303

Even though boiling water reactors (BWRs) are not susceptible to the kind of incident known as pressurized thermal shock that must be considered in the design and operation of pressurized water reactors, BWR reactor pressure vessels (RPVs) have experienced higher than expected embrittlement caused by fast neutron irradiation This has required the vessel to be at a higher temperature than originally projected before the plant can be taken to power operation In addition, many BWR plants have received exemption from the 10-year volumetric nondestructive evaluation (NDE) of the vessel as required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," because NDE access is severely restricted Since many RPV welds have not been examined after being placed in service and the potential for service-induced flaws exists, regulatory authorities are looking closely at examination relief requests BWR reactor vessel examination coverage is typically limited by plant design Most BWR plants were designed when inservice examination codes were in the early stages of development, and very little consideration was give to designing for NDE access Consequently, there is restricted access for many areas of the RPV Since an increase in examination requirements has been placed in ASME B&PV Code Section XI in these areas, efforts have begun on a thorough analysis of the vessel weld volumes examined during inservice examination and an evaluation of possibility expanding the RPV examination coverage Because of these concerns, an investigation of the accessibility of the reactor vessel for NDE was performed to define the present status and to determine the improvements in coverage that can be accomplished in the near future 7 refs , 9 figs , 4 tabs

5
(EUR-12400)
Analysis of data from the Pericles and reflex experiments using the Codes Trac-PF1/MOD1 and QFLOOD. Thomas, R M (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 217p NTIS (US Sales Only), PC A10/MF A01

The computer programs TRAC-FP1/MOD1 and QFLOOD have been used to analyse data obtained from two reflood rigs PERICLES, a 7 - 51 bundle arranged to investigate the chimney effect and REFLEX, a single heated tube TRAC produced poor predictions for PERICLES, the calculated temperature history curves at the 2 03 m elevation differing markedly from experiment TRAC predictions for the REFLEX base case agreed quite well with experiment, but for a second REFLEX test, at higher inlet water flowrate, TRAC greatly overpredicted the quench front speed QFLOOD also performed badly against PERICLES, quench time being overpredicted by more than 50% A number of sensitivity studies were carried out in order to establish the source of the error in the modelling Several possible explanations were investigated, but definite conclusions could not be drawn QFLOOD predictions for REFLEX were generally satisfactory

6
(EUR-12403)
Assessment of RELAP5/MOD2 and RELAP5/MOD1-EUR codes on the basis of LOBI-MOD2 test results. D Auria F, Mazzini, M, Oriolo, F, Galassi, G M (Commission of the European Communities, Luxembourg (Luxembourg)) Oct 1989 100p NTIS (US Sales Only), PC A05/MF A01

The present report deals with an overview of the application of RELAP5/MOD2 and RELAP5/MOD1-EUR codes to tests performed in the LOBI/MOD2 facility The work has been carried out in the frame of a contract between Dipartimento di Costruzioni Meccaniche e Nucleari (DCMN) of Pisa University and CEC The Universities of Roma, Pisa, Bologna and Palermo and the Polytechnic of Torino performed the post-test analysis of the LOBI experiment under the supervision of DCMN In the report the main outcomes from the analysis of the LOBI experiments are given with the attempt to identify deficiencies in the modelling capabilities of the used codes

7
(EUR-12404)
The application of Cathare 1 V1.3 to LOBI small break Loca experiments and a comparison with RELAP5/MOD2. D'Auria, F, Galassi, G M (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 54p NTIS (US Sales Only), PC A04/MF A01

The paper presents an overview of the application of CATHARE V1 3 to LOBI Small Break LOCA tests, performed at Dipartimento di Costruzioni Meccaniche e Nucleari of Pisa University In particular the development of a new nodalization of LOBI facility is discussed along with the analysis of tests A2 81 (1% CL break) A1-83 (10% CL break) and A1-84 (10% HL break) In the second part of the paper, uncertainties are outlined which are typical of the analysis of experiments in integral test facilities Finally, on the basis of the application of RELAP5/MOD2 to the analysis of test A2-81, a judgement is given about the behaviour of the two codes emphasizing the related advantages and disadvantages

8
(EUR-12405)
Assessment of the system code DRUFAN/ATHLET using results of LOBI tests. Burwell, J M, Kirmse, R E, Kyncl, M, Malhotra, P K (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 376p NTIS (US Sales Only), PC A17/MF A01

Four post-test analyses have been performed by GRS within the Shared Cost Action Programme (SCAP) sponsored by the Commission of the European Communities (contract 3015-86-07 EL ISP D) and by the Bundesminister fuer Forschung und Technologie of the Federal Republic of Germany (Research project RS 739) The four tests were mutually selected by the contractors (CEA, GRS, IKE, Univ Pisa) of activity No 3 and by the project organizer Some of the tests were selected to be analyzed by more than one participant in order to allow comparison between analytical results obtained with different codes or obtained by different code-users DRUFAN/ATHLET verification analyses were performed by IKE too The four tests selected for the GRS activity are - A2-77A (Natural Circulation Test), Analysis with ATHLET - A1-76 (Steam Generator Performance Test), Analysis with DRUFAN - BL-01 (Intermediate Leak), Analysis with ATHLET - A2-81 (Small Leak), Analysis with ATHLET This final report contains the results of the four post test analysis including the comparison between measured and calculated quantities and the description of the applied codes, the selected model of the LOBI facility and the conclusions drawn for the improvement of the codes models

9
(EUR-12406)
Hydraulic behaviour of a partially uncovered core. Fischer, K, Hafner, W (Commission of the European Communities, Luxembourg (Luxembourg)) Oct 1989 85p NTIS (US Sales Only) PC A05/MF A01

A critical review of experimental data and theoretical models relevant to the thermohydraulic processes in a partially uncovered core has been performed Presently available optimized thermohydraulic codes should be able to predict swell level elevations within an error band of ± 0 5 m Rod temperature rising velocities could be predicted within an error bandwidth of ± 10%, provided the correct rod heat capacity is given A general statement about the accuracy of predicted rod temperatures is not possible because the errors increase with simulation time Highest errors are expected for long transients with low heating rates and low steam velocities As a result, three areas for additional research are suggested - a high-pressure test at 120 bar to complete the void correlation data base, a low steam flow - low power experiment to improve heat transfer correlations, - a numerical investigation of three dimensional effects in the reactor core with unequally heated rod bundles For the present state of 1-dimensional experiments and models, suggestions for a satisfactory modeling have been derived The suggested further work could improve the modelling capabilities and the code reliability for some limiting cases like high pressure boil-off, low power long term steam cooling, and unequal heating of neighbouring bundles considerably

10
(EUR-12407)
Analysis of experiments performed at University of Hannover with Relap5/Mod2 and Cathare codes on fluid dynamic effects in the fuel element top nozzle area during refilling and reflooding. Ambrosini, W, D'Auria, F, Di Marco, P, Fantappie, G,

Giot, G , Emmerechts, D , Seynhaeve, J M , Zhang, J (Commission of the European Communities, Luxembourg (Luxembourg)) Nov 1989 144p NTIS (US Sales Only), PC A07/MF A01

The experimental data of flooding and CCFL in the fuel element top nozzle area collected at the University of Hannover have been analyzed with RELAP5/MOD2 and CATHARE V 1 3 codes Preliminary sensitivity calculations have been performed to evaluate the influence of various parameters and code options on the results However, an a priori rational assessment procedure has been performed for those parameters non specific in experimental data (e g energy loss coefficients in flow restrictions) This procedure is based on single phase flow pressure drops and no further tuning has been performed to fit experimental data The reported experimental data and some others demonstrate the complex relation ship among the involved physical quantities (film thickness pressure drop etc) even in a simple geometrical condition with well defined boundary conditions In the application of the two advanced codes to the selected CCFL experiments it appears that sophisticated models do not simulate satisfactorily the measured phenomena mainly when situations similar to nuclear reactors are dealt with (rod bundles) This result should be evaluated considering that - dimensional phenomena occurring in flooding experiments are not well reproducible with one dimensional models implemented in the two codes, - a rational and reproducible procedure has been used to fix some boundary conditions (K-tuning), there is the evidence that more tuning can be used to get results closer to the experimental ones in each specific situation, - the uncertainty bands in measured experimental results are not (entirely) specified The work performed demonstrated that further applications to CCFL experiments of present codes appear to be unuseful New models should be tested and implemented before any attempt to reproduce CCFL in experimental facilities by system codes

11
(EUR-12578 pp 311 325)
Exploratory trend and pattern analysis of Caorso plant through the Tenda program. Barsanti P , Tabellini, M (ENEA Rome (IT)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF 8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The eleven years of operating experience of Caorso NPP supply a consistent and homogeneous data set of events involving safety-related systems and components The above data, periodically transmitted to the Italian regulatory body (ENEA/DISP) by the utility (ENEL), as required by the italian rules and regulations, are collected into a computerised Data Bank (SEME) A PC software package (TENDA), which uses, as input, the codes from the SEME Data Bank, was set-up inside ENEA/DISP, with the aim to perform automatic Trend and pattern Analysis Graphic software was also utilized for a more self-explaining pre sentation of the results They are being utilized as input for subsequent studies related to other plants (e g PRA and living PRA as well) Plans for the utiliza tion of the TENDA program for new plants and in the conventional area are now under consideration

12
(EUR-12578, pp 327-338)
Trend analysis on component level using PSA. Carlsson, L , Nilsson, T (Swedish Nuclear Power Inspectorate Stockholm (SE)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The paper outlines the different activities in Sweden to utilize PSA for monitoring safety of Swedish nuclear power plants It is a systematic use of existing data bases Ongoing research is developing the necessary statistical tools The paper is also questioning what should be monitored and for what purpose At present the data will be given in data books for transients component failures and repair rates

13
(EUR-12578, pp 339 350)
Benefits and limitations in the use of performance indicators. Nobile, M , Ceccantini, M (Ente Nazionale per l'Energia Elettrica Rome, (IT)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

In the paper, some Caorso NPP performance indicators for 1982 through 1986 are shown according to the definition of the Institute of Nuclear Power Operations (INPO) The comparison of the Caorso NPP performance indicators with those of similar American plants stresses an analogous behaviour for all the indicators but two (Industrial Safety Lost-Time Accident Rate and Diesel Generator Unavailability), for which there is a very significant difference A subsequent analysis enhances, however, that this difference cannot be attributed to component performance or personnel management politics but essentially to design choices and/or to peculiar social and economical context, if these factors are taken into consideration, the difference between the performance indicators of the Caorso plant and those of American plants is explained quite satisfactorily This demonstrates, that even if the usefulness of the performance indicators is confirmed, a careful analysis of all the possible implications is necessary before reaching any final conclusion

14
(EUR-12578, pp 363-377)
The backfitting process and its verification. Del Nero, G , Grimaldi, G 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF 8904156- CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of op erational data from nuclear power plants

Backfitting of plants in operation is based on - compliance with new standards and regulations, - lessons learned from operating experience This goal can be more effectively achieved on the basis of a valid methodology of analysis and a consistent process of collection, storage and retrieval of the operating data The general backfitting problem, the verification process and the utilization of TPA as mean to assess backfitting are illustrated The results of the analyses performed on Caorso plant are presented as well, using some specially designed software tools Management more than hardware problems are focused Some general conclusions are then presented as final results of the whole work

15
(EUR-12578, pp 419-435)
Reliability analysis of the diesel generators. Curcuruto, S, Grimaldi, G (ENEA Rome, (IT)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159).

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The Operating Experience of the Diesel generators of the Italian Nuclear Power Plants has been analysed, in order to evaluate their quality level in comparison with the requirements defined in the design and to identify possible improving measures to be implemented both on operating and under construction plants The collected data have been classified and elaborated, with the purpose to evaluate availability on demand and reliability in operation for each diesel A comparison between the calculated reliability parameters and the corresponding international ones was also performed Recurring failure modes were specifically analysed In addition, an analysis of homogeneity for the diesels of the same plant and of all the plants is reported as well In such a way, some critical subsystems of DGs has been identified Moreover hardware modifications and surveillance program improvements have been found out, such to ensure better performance of the on-site electric power system

16
(EUR-12578, pp 449 460)
Load factor trends in light water reactor units. Lehtinen, E A (Valtion Teknillinen Tutkimuskeskus, Espoo (FI)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The Technical Research Centre of Finland follows up and analyses nuclear power plant availability performances worldwide The results of a trend study for the load factors of the LWR units have been updated to the end of 1987 The whole operating history, in the sense of the annual and cumulative load factors achieved by all the Western commercial LWR units until the end of 1987, has been taken into consideration Some trends in the load factors have been identified by using an exponential regression model developed The LWR units form quite an inhomogeneous population with respect to their age, technical characteristics, site country as well as cumulative load factors achieved The cumulative load factors achieved by all the LWR units until the end of 1987 are presented individually in the scattergrams

17
(EUR-12578, pp 475-490)
10 Years of operating experience of the valves in the safety systems on Caorso plant. Curcuruto, S, Pasquini, M (ENEA, Rome, (IT)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The Operating Experience (O E) of the valves in the safety related systems on Caorso plant has been analysed The valves have been grouped according to system, type and manufacturer All the data on the failures have been respectively drawn out by the O E data bank and, in some cases, they have been integrated by informations collected directly on the plant The events and the relevant causes have been analysed, particularly taking into account the repetitive events Most of the failures were discovered during the surveillance tests, giving a positive indication of the effectiveness of the periodic test program It was also that concluded hardware problems caused more failures than human errors both during operation and maintenance Abnormal distributions of failures on the valves and on their components have been found out Weak components both mechanical and electrical and pertinent corrective measures have been identified, aimed to eliminate the recurring failure modes

18
(EUR-12578, pp 501-511)
Long term trend analysis of emergency power diesel generator reliability in german nuclear power plants. Kotthoff, K, Maqua, M (Gesellschaft fuer Reaktorsicherheit mbH (GRS) Koeln, (DE)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

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The paper deals with a long-term investigation on the availability of diesel generators This investigation has been performed in two steps in 1980/81 and 1988/89 It is based on the operating experiences of a total of 110 diesel generators in 20 German NPP's The overall probability of diesel failure during start and short-time operation amounts to about 5E-3/demand Compared to the result of the first part of the investigation (8E-3/demand) there has been some further improvement of diesel generator performance in recent years The upper limit calculated for the probability of common mode failures (about 6E-4/demand) is approximately one order of magnitude smaller The influence of various parameters on the failure probability has been discussed A statistically significant dependence could not be identified

19
(EUR-12578, pp. 513-528)
Trend and pattern analysis of human performance problems at the swedish nuclear power plants. Bento, J P (Studsvik Nuclear, Nykoeping (SE)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The last six years of operation of all Swedish nuclear power plants have been studied with respect to human performance problems by analysing all scrams and licensee event reports (LERs) The present paper is an updated version of a previous report to which the analysis results of the year 1988 s events have been added The study covers 197 scrams and 1759 LERs As general results, 38% of the scrams and 27% of the LERs, as an average for the years 1983-1988, are caused by human performance problems Among the items studied, emphasis has been put on the analysis of the causal categories involved in human performance problems resulting in plant events The most significant causal categories appear to be Work organization, Work place ergonomics, Procedures not followed, Training and Human variability The trend and pattern of the dominating causal categories are discussed

20
(JAERI-M-90-109)
Studies of neutronics calculation of high conversion light water reactor. Akie, Hiroshi, Okumura, Keisuke, Takano, Hideki, Ishiguro, Yukio (Japan Atomic Energy Research Inst Tokyo (Japan)) Jul 1990 53p NTIS (US Sales Only), PC A04/MF A01 Order Number DE91723320

The studies of data and methods have been made to enhance reliability for neutronics calculation of high conversion light water reactor (HCLWR) Treatments of heavy resonant nuclides and fission products in the calculation of resonance absorption and in burnup calculation have been mainly investigated The purpose of the investigations is to accurately predict important nuclear characteristics such as coolant void reactivity and burnup reactivity swing At the same time, experimental analysis and benchmark calculation have also been performed to test the accuracy of data and methods The JAERI standard code system for thermal reactor design and analysis, SRAC, has been improved on the basis of the present studies (author)

21
(LA-11965-MS)
Neutron collar calibration and evaluation for assay of LWR fuel assemblies containing burnable neutron absorbers. Henriksen, P W (ed), Menlove, H O , Stewart, J E , Qiao, S Z , Wenz, T R , Verrecchia, G P D (Los Alamos National Lab , NM (USA)) Nov 1990 Contract W-7405-ENG-36 48p (ISPO-323) NTIS, PC A04/MF A01 - OSTI, GPO Dep Order Number DE91001718

The neutron coincidence collar is used to verify the uranium content in light water reactor fuel assemblies An AmLi neutron source actively interrogates the fuel assembly to measure the ^{235}U content and the ^{238}U content can be verified from a passive neutron coincidence measurement This report gives the collar calibration data for pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assemblies both with and without cadmium liners Calibration curves and correction factors are presented for neutron absorbers (burnable poisons) and various fuel assembly sizes The data were collected using the Los Alamos BWR and PWR test assemblies as well as fuel assemblies from several fuel fabrication facilities 11 refs , 15 figs , 14 tabs

22
(NUREG/CR-4219-Vol 6-No 2)
Heavy-Section Steel Technology Program. Pennell, W E (Nuclear Regulatory Commission, Washington, DC (USA) Div of Engineering, Oak Ridge National Lab , TN (USA)) Aug 1990 Contract AC05-84OR21400 73p (ORNL/TM-9593-Vol 6-No 2) NTIS, PC A05/MF A01 - GPO, OSTI, INIS

During the current report period, the evaluation of materials properties advances on the pressurized-thermal-shock (PTS) analysis was continued This effort was supported by the development of analytical models for the dynamic analysis of the wide-plate crack-arrest specimens and the development of a stub-panel specimen for use in an extended crack-arrest testing program Studies of the impact of ductile tearing on the PTS analysis result have been initiated Analytical studies of constraint effects continued Development continued on a high-strain-rate testing technique, and development of strain-rate-dependent inelastic materials models continued for application to the analysis of dynamic fracture-toughness tests A review of the history of the LUS toughness issue was completed, and a draft of the report was forwarded to NRC Development of crack-arrest testing technology, and irradiation effect studies were continued Preparation for the study of the effects of aging on reactor system structural materials was initiated in this reporting period with an emphasis on a detailed assessment of the capability of existing facilities at ORNL for conducting the required materials examinations and tests

23
(NUREG/CR-5314-Vol 3)
Life assessment procedures for major LWR [light water reactor] components: Cast stainless steel components. Jaske, C E , Shah, V N (Nuclear Regulatory Commission, Washington, DC (USA) Div of Engineering, EG and G Idaho, Inc Idaho Falls, ID (USA)) Oct 1990 Contract AC07-76ID01570 49p (EGG-2562-Vol 3) NTIS, PC A04/MF A01 - GPO, OSTI, INIS

This report presents a procedure for estimating the current condition and residual life of safety-related cast stainless steel components in light water reactors (LWRs) The procedure accounts for loss of fracture toughness caused by thermal embrittlement and includes the following a review of design and fabrication records, inservice inspection records, and operating history, a fracture mechanics evaluation to determine the required toughness at end-of-life using worst-loads and worst-flaw indications, current and future toughness estimates, and criteria regarding continued service, repair, or replacement of the component being evaluated The report discusses the available Charpy V-notch impact energy, fracture toughness, tensile strength, fatigue resistance, and fatigue-crack growth data, and presents two methods for assessing the degree of thermal embrittlement metallurgical evaluation and analytical modeling of inservice degradation 74 refs , 21 figs , 5 tabs

24
(NUREG/CR-5605)
LAPUR benchmark against in-phase and out-of-phase stability tests. March-Leuba, J (Nuclear Regulatory Commission, Washington, DC (USA) Div of Systems Technology, Oak Ridge National Lab , TN (USA)) Oct 1990 Contract AC05-84OR21400

27p (ORNL/TM-11621) NTIS PC A03/MF A01 - GPO, OSTI INIS

This paper documents a benchmark of the LAPUR code vs experimental stability data collected during startup testing of the Oskarshamn-3 reactor The data consist of decay ratios and natural frequencies of oscillation measured under several reactor operating conditions Satisfactory agreement was found between the measured decay and those calculated by LAPUR for both the in-phase and out-of-phase instability modes The largest error (0 11) in the calculated decay ratio corresponded to test point 3, which corresponds to a very stable condition for which experimental decay ratio measurements are difficult 4 refs , 8 tabs

25

(ORNL/TM 11556)

Characteristics Data Base. Lewis E D Moore R S (Oak Ridge National Lab TN (USA) Automated Sciences Group Inc Oak Ridge, TN (USA)) Aug 1990 Contract AC05-84OR21400 ,AC05 86OR21642 86p NTIS, PC A04/MF A01, OSTI, INIS, GPO Dep Order Number DE91001023

The LWR Serial Numbers Database System (SNDB) contains detailed data about individual historically discharged LWR spent fuel assemblies This data includes the reactor where used, the year the assemblies were discharged the pool where they are currently stored, assembly type, burnup, weight, enrichment, and an estimate of their radiological properties This information is distributed on floppy disks to users in the nuclear industry to assist in planning for the permanent nuclear waste repository This document describes the design and development of the SNDB It provides a complete description of the file structures and an outline of the major code modules It serves as a reference for a programmer maintaining the system, or for others interested in the technical detail of this database This is the initial version of the SNDB It contains historical data through December 31, 1987, obtained from the Energy Information Administration (EIA) EIA obtains the data from the utility companies via the RW-859 Survey Form It evaluates and standardizes the data and distributes the resulting batch level database as a large file on magnetic tape The Characteristics Data Base obtains this database for use in the LWR Quantities Data Base Additionally, the CDB obtains the individual assembly level detail from EIA for use in the SNDB While the Quantities Data Base retains only the level of detail necessary for its reporting, the SNDB does retain and use the batch level data to assist in the identification of a particular assembly serial number We expect to update the SNDB on an annual basis, as new historical data becomes available

26

(SAND-90-2716C)

Results of the DF-4 BWR [boiling water reactor] control blade-channel box test. Gauntt, R O , Gasser R D (Sandia National Labs , Albuquerque NM (USA)) Oct 1990 Contract AC04-76DP00789 15p (CONF-9010185-7) NTIS PC A03/MF A01 - OSTI GPO Dep Order Number DE91002423

From 18 water reactor safety information meeting Gaithersburg MD (USA) (22-24 Oct 1990)

The DF-4 in-pile fuel damage experiment investigated the behavior of boiling water reactor (BWR) fuel canisters and control blades in the high temperature environment of an unrecovered reactor accident This experiment, which was carried out in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories, was performed under the USNRC s internationally sponsored severe fuel damage (SFD) program The DF-4 test is described herein and results from the experiment are presented Important findings from the DF-4 test include the low temperature melting of the stainless steel control blade caused by reaction with the B_4C, and the subsequent low temperature attack of the Zr-4 channel box by the relocating molten blade components Hydrogen generation was found to continue throughout the experiment, diminishing slightly following the relocation of molten oxidizing zircaloy to the lower extreme of the test bundle A large blockage which was formed from this material continued to oxidize while steam was being fed into the the test bundle The results of this test have provided information on the initial stages of core melt progression in BWR geometry involving the heatup and cladding oxidation stages of a severe accident and terminating at the point of melting and relocation of the metallic core components The information is useful in modeling melt progression in BWR core geometry, and provides engineering insight into the key phenomena controlling these processes 12 refs , 12 figs

27

(WSRC-MS-90-38)

A failure probability estimate of Type 304 stainless steel piping. Daugherty, W L , Awadalla, N G , Sindelar, R L , Mehta, H S (Westinghouse Savannah River Co , Aiken, SC (USA)) [1990] Contract AC09-89SR18035 7p (CONF-9010151-3) NTIS, PC A02/MF A01, OSTI, INIS, GPO Dep Order Number DE91004299

From International topical meeting on the safety, status, and future of non-commercial reactors and irradiation facilities, Boise ID (USA) (4 Oct 1990)

The large break frequency resulting from intergranular stress corrosion cracking in the main circulation piping of the Savannah River Site (SRS) production reactors has been estimated Four factors are developed to describe the likelihood that a crack exists that is not identified by ultrasonic inspection and that grows to instability prior to becoming through-wall and being detected by the ensuing leakage The estimated large break frequency is 3 4 $\times$ 10^{-8} per reactor year This result compares favorably to similar estimates made for commercial boiling water reactors 9 refs , 8 figs

28

Berechnung der Nachzerfallsleistung der Kernbrennstoffe von Leichtwasserreaktoren. Nichtrezyklierte Kernbrennstoffe (Decay heat power in nuclear fuels of light water reactors. Non-recycled nuclear fuels). Berlin (Germany, F R), Beuth (May 1990) 12p (In German) (DIN-25463(pt 1))

The standard supplies the basis for the calculation of the decay heat power of non-recycled nuclear fuel The following parts are taken into account the contribution of fission products from nuclear fission, the contribution of the actinides, the contribution of isotopes which are produced, by neutron capture in fission products It shows the local production of decay heat power relative to the thermal fuel power during operation The process of calculation standardized here has the advantage of calculating the decay heat power with an accuracy comparable to summation programs, without needing expensive computer programs and extensive data libraries (orig /HP)

29
Berechnung der Nachzerfallsleistung der Kernbrennstoffe von Leichtwasserreaktoren. Nichtrezyklierte Kernbrennstoffe. Dokumentation und Erlaeuterungen (Decay heat power in nuclear fuels of light water reactors. Non-recycled nuclear fuels; documentation and illustration). Berlin (Germany, F R), Beuth (May 1990) 28p (In German) (DIN-25463(pt 1,suppl))

This additional sheet contains information for DIN 25 463 Part 1 but no additional standardized regulations (orig)

30
Reaktorsicherheitsbehaelter aus Stahl. T. 4. Wiederkehrende Pruefungen (Steel containment for nuclear reactors. Pt. 4. In-service inspections). Koeln (Germany, F R), Heymanns (1990) 16p (In German) (KTA-3401 4(draft,ed 6/90))

The standard is valid for in-service inspections of steel containments for stationary LWRs including those airlocks construction and transport openings, pipe and cable penetrations which are firmly connected with the containment as well as isolating devices of pipings which have an open end in the containment atmosphere and serve for containment isolation The standard is also valid for containments with a pressure suppression system and an external liner It is to be applied to all in-service inspections following first criticality, including the first repetition of the integral leak rate test (preoperational leak rate test) The aim of the standard is to fix the extend and the intervals for the in-service inspections performed on the containment and its components in order to test its integrity and performance (orig)

31
Komponentenstuetz konstruktionen mit nichtintegralen Anschluessen. T. 1. Komponentenstuetzkonstruktionen mit nichtintegralen Anschluessen fuer Primaerkreiskomponenten in Leichtwasserreaktoren (Component support constructions with non-integral connections. Pt. 1. Component support constructions with non-integral connections and deflection safety books for primary loop components in light water reactors). Koeln (Germany, F R), Heymanns (1990) 115p (In German) (KTA-3205 1(ed 6/90))

The rule has to be applied to component support constructions with non-integral connections and deflection safety books for primary loop components with design temperatures up to 623 K (350 deg C) in LWRs (orig)

32
Impact of degraded cooling water system performance upon power plant operations. Bogard, T , Arnold, E , Cefola, G pp 943-954 of Power-gen 1989 Conference papers, Volumes V and VI Houston, TX (US), Power-Gen (1989) 413p (CONF-891217-)

From POWER-GEN '89 2nd conference and exhibition for the power generation industries, New Orleans, LA (USA) (5-7 Dec 1989)

A number of recent events have focused attention on the performance of cooling water systems at power plants In various region of the country, the unusual weather and temperatures occurring during the past several summers have led to abnormally high ultimate heat sink temperatures In addition, the industry is addressing the effect of component aging upon cooling water system performance The Nuclear Regulatory Commission (NRC) has issued generic letter 89-13, which summarizes their concerns regarding performance degradation of cooling water systems As a result of the meteorological phenomenon and to identify potential system degradation, equipment reviews and analyses of cooling water systems and serviced components are necessary These efforts determine if any degraded system or equipment performance exists and the consequential affect on the serviced components This paper reviews the factors leading to service water system degradation, the methods to quantify reduced system capability, and the evaluation of component performance with reduced cooling water systems flow or increased cooling water temperature

33
Service water corrosion control at River Bend Station. Hamilton, J R , Brice, T O , Sankovich, M F , Morris, J G pp 1035-1048 of Power-gen 1989 Conference papers, Volumes V and VI Houston, TX (US), Power-Gen (1989) 413p (CONF-891217-)

From POWER-GEN '89 2nd conference and exhibition for the power generation industries; New Orleans, LA (USA) (5-7 Dec 1989).

Service water systems are demanding more engineering, operations, and maintenance attention than ever before. Problems affecting service water systems have become more complex and more frequent due to increased environmental restrictions and unique concerns such as asiatic clams Proven corrosion inhibitors like chromates are no longer permitted for discharge to the environment Treatment program choices available to control biofouling, reduce corrosion, and prevent deposition and fouling are limited The result is treatment programs which are less effective and which create an environment for Microbiological Influenced Corrosion (MIC) Once accelerated corrosion begins, a complex program of system design, water treatment, corrosion/deposition monitoring, inspection, repair, and operation may be necessary to restore the system Such a program, necessitated by initial operating experience, is now in place at River Bend Station Our experience is not unique Both the Electric Power Research Institute and the Nuclear Regulatory Commission have initiated programs to address service water system problems, which include accelerated corrosion, deposition and fouling, degradation of heat exchanger performance and MIC

34
Cracked shaft diagnosis and detection on reactor recirculation pumps at Grand Gulf Nuclear Station. Allen, J W , Bohanick, J S pp 1021-1034 of Power-gen 1989 Conference papers, Volumes V and VI Houston, TX (US), Power-Gen (1989) 413p (CONF-891217-)

From POWER-GEN '89 2nd conference and exhibition for the power generation industries, New Orleans, LA (USA) (5-7 Dec 1989)

By utilizing state-of-the-art analysis techniques, instrumentation and vibration information, a cracked shaft was hypothesized and confirmed for one of two Reactor Recirculation Pumps at Grand Gulf Nuclear Station (GGNS) This experience confirmed many previous observations and expected behavior patterns for cracked shafts and, not unexpectedly, provided another unique set of data which can be added to the data base being developed for cracked shaft occurrences

35
Comparison of k_∞ void coefficient results from LWHCR experiments with different moderation ratios. Boehme, R , Berger, H D , Chawla, R , Hager, H , Pelloni, S , Seiler, R (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp I 45 I 53 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409–)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The series of physics experiments being carried out in the PROTEUS reactor in Switzerland has been extended to the investigation of a Pu-fueled light water high converter reactor (LWHCR) lattice with a moderator-to-fuel volume ratio of 0 95 The analysis of the new measurements in the wider LWHCR test lattice indicates a dependence of the ratio of calculated to measured reaction rate ratios on the degree of moderation Due to compensating errors in the reaction rate balance the predictions of k_∞ appear consistent The dependence of the observed trends on nuclear data and methods has revealed some deficiencies in the calculation of resonance absorption and in the nuclear data at higher neutron energies

36
Advances in the analysis of the NEACRP high conversion LWR benchmark problems. Bernnat, W , Ishiguro, Y , Sartori, E , Takano, M , Stepanek, J (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp I 54 I 63 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409–)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The Reactor Physics Committee of NEA (NEACRP) has sponsored an international benchmark of tight lattice cell burn-up calculations proposed by JAERI to validate the computational methods for High Conversion Light Water Reactor applications The activities and results of the HCLWR Burn-up Working Group consisting of participants from 18 institutions of 8 countries are summarized Two lattices with moderator/fuel ratio of 0 6 and 1 1 as well as the core 1-6 configurations of the PROTEUS Phase I experiments were analyzed with different production codes and one continuous Monte Carlo code Several iterations on the benchmark calculations were required to resolve some of the large discrepancies found in the first set of results Results obtained with codes and data reflecting the state-of-the-art indicate that agreement between the solutions has a quality comparable to the one observed in standard LWR-benchmark solutions

37
Venus international programme (VIP) a nuclear data package for LWR Pu recycle. Charlier, A , Basselier, J , Leenders, L (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 65-VI 72 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409–)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Large-scale MOX irradiation campaigns are now considered in various industrial countries Consequently, there is a need to develop, to improve and to validate nuclear computer codes The objective of the VIP programme is to constitute a complete set of experimental measurements performed with UO_2 and MOX fuel rods in order to provide an extensive nuclear data base for the development and the validation of nuclear calculation methods for MOX fuels used in LWRs A VIP programme will be devoted to Pu recycle in PWRs and another one in BWRs The VIP programme is starting now

38
HECTR assessment of some HDR experiments. Pong, L T *Nuclear Engineering and Design (Netherlands),* 121 No 1 25 38 (Jul 1990) (CONF-890819–)

From ASME/AIChE national heat transfer conference Philadelphia, PA (USA) (6-9 Aug 1989)

The HECTR (Hydrogen Event Containment Transient Response) code was assessed against two large-scale thermal-hydraulic and hydrogen mixing tests (Tests 31 1 and 31 5) in the German Heiss Dampf Reaktor (HDR) decommissioned reactor facility HECTR is a lumped volume code with models for the transport, mixing, and combustion of hydrogen HECTR was modified to include an isentropic expansion process to describe two-phase water blowdown and a simple model to calculate an effective compartment velocity in each time step The specific objective of this work is to assess the transport and mixing models in the HECTR code (orig)

39
On-screen design makes updating easy at Japanese BWR project. Neeley, G R , Bridenstine, D E , Yoshinaga, T , Hayashi, K *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 53-55 (Jul 1990)

The complex problems of the design and updating of balance of plant have been solved at a new Boiling Water Reactor project in Japan by the use of advanced three-dimensional computer aided design The system is described (author)

40
Making progress with Individual Plant Examinations in the United States. Cave, L *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 431, 37-38 (Jun 1990)

The US Nuclear Regulatory Commission's NRC's Individual Plant Examination programme is part of the larger Severe Accident Programme which was initiated in 1985 The objectives are to investigate, for all US LWRS in operation, the adequacy of the systems for preventing damage to the reactor core in fault conditions, the systems for protecting the containment, if severe core damage occurs, the emergency procedures and associated instrumentation and control for arresting fault sequences leading towards core damage or containment failure (author)

41
Mechanized wall thickness inspection measures up to its cost. Edelmann, X , Gribi, M *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 54-55 (May 1990)

Mechanized pipe wall thickness measurement with ultrasonics has opened up a new dimension in plant surveillance, in particular when dealing with the problem of erosion-corrosion in Pressurized and Boiling Water Reactors Although the costs are higher than with manual measurement, Gebruder Sulzer has found that the performance and reliability of the P-Scan system outweighs this disadvantage (author)

42
Alteration of reactor installation (alteration of No.3 reactor facilities) in Fukushima No.2 Nuclear Power Station, Tokyo Electric Power Co., Inc. (Report). (Nuclear Safety Commission, Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 13-15 (Mar 1990) (In Japanese)

The Nuclear Safety Commission received the report on December 15, 1989 from the Committee on Examination of Nuclear Reactor Safety on this alteration about which the instruction to the Committee had been made on November 9 1989 and after the deliberation, it presented the report to the Minister of International Trade and Industry It was recognized as the result of examination that the technical capability of the applicant is appropriate It was judged as the results of investigation and deliberation that the safety after this alteration of reactor installation can be ensured The content of this alteration of installation is, at the time of third fuel replacement, to use the replacement fuel, of which the average enrichment degree is lowered to about 2 2 mt %, in addition to the conventional replacement fuel of 3 0 wt % enrichment because it is necessary to charge more new fuel than usual case The mechanical design and the nuclear design of the replacement fuel were examined, and it was confirmed that the safety design is appropriate (K I)

43
Alteration of reactor installation (alteration of No.1, No.2, No.3 and No.4 reactor facilities) in Hamaoka Nuclear Power Station, Chubu Electric Power Co., Inc. (Report). (Nuclear Safety Commission, Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 15-17 (Mar 1990) (In Japanese)

On this alteration, about which the inquiry had been received on November 6, 1989, the Nuclear Safety Commission carried out prudent deliberation, and presented the report to the Minister of International Trade and Industry It was recognized as the result of examination that the technical capability of the applicant is appropriate It was judged as the result of examination that the safety after this alteration of the installation of reactor facilities can be ensured The main contents of alteration concerning this application are as follows The start-up region monitors are adopted in place of neutron source region monitors and intermediate region monitors in No 1, 2 and 4 reactors Hafnium type control rods are partially adopted in No 1, 2 and 3 reactors The type of main steam isolation valves is changed in No 4 reactor The capacity of a spent resin storage tank for No 4 reactor purification system is changed The type of uninterruptible AC power source equipments is changed in No 1 and 2 reactors The safety design of reactor facilities, the dose equivalent that the public in surroundings receive, the analysis of abnormal transient change in operation, the analysis of accidents, and the analysis of credible accidents for the evaluation of location were examined (K I)

44
Results of 10th regular inspection of No.1 plant in Hamaoka Nuclear Power Station, Chubu Electric Power Co., Inc. (Agency of Natural Resources and Energy, Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 22-23 (Mar 1990) (In Japanese)

The 10th regular inspection of No 1 plant in Hamaoka Nuclear Power Station was carried out from June 18, 1988 to August 24, 1989 The parallel operation was resumed on July 17, 1989, 395 days after the parallel off The facilities which were the object of inspection were the reactor proper, reactor cooling system measurement and control system fuel facilities, radiation control facilities, waste facilities, reactor containment installation and emergency power generation system On these facilities which were the object of inspection the appearance, disassembling, leak, function, performance and other inspections were carried out As the result, in one in-core monitor housing, slight water oozing-out was observed but other abnormality was not found The works related to this regular inspection were accomplished within the range of the limit of dose equivalent based on the relevant laws The main reconstruction works carried out during the period of this regular inspection were the replacement of the exhaust gas condenser, the replacement of piping at the penetration of the reactor containment vessel, the adoption of new 8 x 8 zirconium liner fuel, the replacement of the regenerative heat exchanger, the replacement of the exhaust gas-water separator, the repair of the in-core monitor housing and the replacement of submerged bearings for reactor recirculation pumps (K I)

45
Operating experience with new moisture preseparators and special crossunder pipe separators. Improved plant efficiency and availability. *ABB Review (Switzerland),* No 3, 3-10 (1990)

German version published in ABB Tech (1990) (no 3) p 3-10

Construction has begun of the first installations to be based on ABB's new moisture separator reheater concept The installations comprise the moisture preseparator (MOPS), special crossunder pipe separator (SCRUPS) and reheater of advanced design (ROAD) Valuable operating experience has been gained with the MOPS and SCRUPS in recent years Tests performed on SCRUPS prototypes in a Swiss nuclear power plant have pointed to excellent separation efficiency, and the equipment has performed very satisfactorily since being installed five years ago A combined MOPS/SCRUPS for retrofitting in existing steam plants is currently in operation in ten plants and experience with it has been good It solves the problems of moisture separation, improves plant efficiency and stops erosion-corrosion in the crossunder pipe and moisture separator (orig)

46
Boiling water cooled and moderated VK reactor is the optimal one for NPP. Kruzhilin, G N, Anan'ev, E P, Dubrovskij, I S *Ehlektricheskie Stantsii (USSR),* No 10, 18-21 (Oct 1989) (In Russian)

Comparative analysis of safety problems in the RBMK, WWER and VK type reactors is performed The possibility to use the VK type reactors at NPPs instead of the RBMK type reactors in particular which are withdrawn from production after the Chernobyl accident is proved The possibility is confirmed by foreign experience in NPP operation with BWR type reactors and successful operation of the VK-50 reactor in the USSR

47
Neutron irradiation embrittlement of light water reactor pressure vessel steels. Odette, G R, Lucas, G E *Transactions of the American Nuclear*

Society (USA), 60 285-286 (1989) (CONF 891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco CA (USA) (26-30 Nov 1989)

This paper reviews the results of an experimental and theoretical program to develop physically based, predictive models of light water reactor pressure vessel steel embrittlement that the authors have conducted over the past decade Embrittlement was characterized using subsized tensile specimen and microhardness techniques Irradiation-induced microstructure changes have been characterized by small angle neutron scattering (SANS) and atom probe/field ion microscopy techniques Simple diffusion controlled growth models have been applied to irradiation enhanced precipitation with good success

48
Stability of boiling water reactor limit cycle: Bifurcations and chaotic behavior. March-Leuba, J *Transactions of the American Nuclear Society (USA),* 60 345-346 (1989) (CONF-891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The work presented in this paper deals with the stability of the boiling water reactor (BWR) limit cycle Numerical simulations have shown that as a parameter controlling the reactor's linear instability is increased, the limit cycle may become unstable and degenerate, through a cascade of period-doubling pitchfork bifurcations, into a chaotic (i e , aperiodic) sequence of power oscillations In this regime, the power oscillations are aperiodic but remain bounded by a strange attractor It appears however that the oscillation amplitude (i e , peak power) is increased by this phenomenon This effect can be observed which shows a clear discontinuity in the derivative of the peak power at the bifurcation points Thus, the bifurcated limit cycle exhibits larger peak powers than extrapolation of the nonbifurcated limit cycle would predict Thus, the understanding of the bifurcation phenomenon in BWRs is relevant to accident analysis where large peak powers may affect fuel integrity

49
Radiolysis gas measurements in the water-steam circuit of a boiling water reactor (BWR) plant. Staudt, U , Ruehle, W , Hoffmann, W *VGB Kraftwerkstechnik (Germany, F R),* 70 No 9, 799 802 (Sep 1990) (In German)

Hydrogen- and oxygen measurements have been taken at various points in the water-steam circuit of Philippsburg 1 BWR Plant in order to obtain a view of the net production of these gases and their behaviour in the system components which admit steam Important aspects of the investigations were therefore possible radiolysis gas enrichments on components through which there is no or inadequate throughflow and the determination of distribution balances between steam- and condensate phases Sampling points have been inserted and tested for this purpose (orig)

50
PLEVIS: plant engineering visual and interactive simulator. Tanaka, Kazuma, Yoshikawa, Eiji, Ohtsuka, Shiroh, Kawakami, Seishiroh *Karyoku Genshiryoku Hatsuden (Thermal and Nuclear Power) (Japan),* 41 No 7, 884-892 (Jul 1990) (In Japanese)

TOSHIBA has devoted much effort since 1974 to the development of nuclear power plant simulators for operator training, such as replica simulators and compact simulators These simulators have special purpose consoles or control room equipment for operation training, and run on one or more process computers Hence they require a large amount of space, e g a special room for installation PLEVIS is a real time plant engineering simulator and operates on a general purpose desk-side engineering workstation (TOSHIBA AS-series) with a high resolution bit mapped display The major objectives of PLEVIS are to further improve training performance and efficiency for operators and engineers and to provide a powerful and cost effective plant engineering tool (author)

51
Improvement for BWR operator training, 3. Noji, Kunio, Toeda, Susumu, Saito, Genhachi, Suzuki, Koichi *Karyoku Genshiryoku Hatsuden (Thermal and Nuclear Power) (Japan),* 41 No 7, 893-898 (Jul 1990) (In Japanese)

BWR Operator Training Center Corporation (BTC) is conducting training for BWR plant operators using Full-scope Simulators There are several courses for individual operators and one training course for shift crew (Family Training Course) in BTC Family Training is carried out by all members of the operating shift-crew BTC has made efforts to improve the Family Training in order to acquire more effective training results and contribute to up grade team performance of all crews This paper describes some items of our efforts towards Family Training improvement (author)

52
Discharge-preventive wall for reactor core meltdown product. Kume Tadashi Furukawa Hideyasu (to Hitachi Ltd Tokyo (Japan)) Japan Patent 2-136789/A/ 25 May 1990 Filed date 18 Nov 1988 3p (In Japanese) JAPIO Also available from INPADOC

In a reactor container comprising a sealed vessel for containing a reactor vessel, a pressure suppression changer filled with coolant and a vent tube connected at one end to the sealed container and dipped at the other end into the coolants in the pressure suppression chamber, the discharge-preventive walls for the reactor core meltdown products are made of concretes or refractory bricks and tightly bonded at the lower portion thereof with steel-reinforced concretes at the bottom of a dry well Thus, the reactor core meltdown products can be accumulated between the reactor pressure vessel pedestal and the discharge-preventive walls for the reactor core meltdown products to avoid direct contact of the reactor core meltdown products with the reactor container Accordingly, destructions caused in a short time by the reactor core meltdown products under assumed severe accidents reactor core meltdown can be prevented (T M)

53
Iron concentration controller in feedwater in nuclear plant. Aizawa, Motohiro, Isaka, Yoshitaka (to Hitachi Ltd , Tokyo (Japan), Hitachi Engineering Co Ltd , Ibaraki (Japan)) Japan Patent 2-134596/A/ 23 May 1990 Filed date 15 Nov 1988 7p (In Japanese) JAPIO Also available from INPADOC

The purpose of the present invention is to prevent chlorine ions from flowing into a reactor when sea water leakage accident should occur in a condenser upon control of Fe concentration in feedwater That is, a sensor is disposed for detecting the leakage of the sea water at the exit of the condenser The controller receives a detection signal as the input and delivers a control signal as the output A control system receives the control signal and actuates valves in bypass systems In view of the above, the electroconductivity or chlorine ion concentration of the condensate, which varies upon occurrence of sea water leakages in the condenser, is detected by the sensor, and then the controller closes a valve dispposed in the bypass systems in a processing device for filtering and desalting the condensates Accordingly, the chlorine ions mixed into the condensates are removed by a desalting device without flowing into the reactor In view of the above, an effect capable of keeping integrity of the plant is obtainable (I S)

54
Fuel assembly. Chuma, Kazuhito (to Toshiba Corp , Kawasaki, Kanagawa (Japan), Nippon Atomic Industry Group Co Ltd , Tokyo (Japan)) Japan Patent 2-129585/A/ 17 May 1990 Filed date 10 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

In a fuel assembly of the present invention the pressure loss in the single phase portion is increased thereby improving the margin of stability to extend the operation area of a BWR type reactor Fuel rods are enhoused in a fuel assembly and each of them is arranged and disposed by spacers An no-heated portion is set below the fuel assembly and spacers are disposed in the no-heated portion In the fuel assembly thus constituted, the pressure loss of the spacers in the lower portion takes a value in the single phase state Accordingly, the pressure loss of the single phase in the reactor core can be increased to thereby stabilize characteristics in the reactor core and the channels As a result, the stability margin in the BWR type reactor can be improved to extend the operation area (I S)

55
Channel box for fuel assembly. Yokobori, Seiichi (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-126188/A/ 15 May 1990 Filed date 4 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

At least one of concave recess, small protrusion and strip-like net is disposed on the inner surface at the upstream of a channel box where spacers are situated With such a constitution, coolants uprising along the inner circumferential wall surface of the channel box while adhering in the form of liquid membranes are guided conforming the curvature of the recess as they come to the position thereof, directed inward and finally involved in the form of droplets into the main stream of the coolants without deposition The detaching speed of droplets depends on the velocity of the accompanying steams and the shape of the concave recess and an optimum separation efficiency can be obtained for the droplets as the speed of the former is increased and the angle of the latter intersecting with the wall is about 90 to 45 deg In this way, the coolants are guided smoothly in the form of droplets to improve the cooling efficiency of each of fuel rods and to certainly improve the critical power of the reactor (T M)

56
Reactor containment. Kawabe, Ryuhei, Yamaki, Rika (to Hitachi Ltd , Tokyo (Japan)) Japan Patent 2-126189/A/ 15 May 1990 Filed date 7 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

A water vessel is disposed and the gas phase portion of the water vessel is connected to a reactor container by a pipeline having a valve disposed at the midway thereof A pipe in communication with external air is extended upwardly from the liquid phase portion to a considerable height so as to resist against the back pressure by a waterhead in the pipeline Accordingly, when the pressure in the container is reduced to a negative level, air passes through the pipeline and uprises through the liquid phase portion in the water vessel in the form of bubbles and then flows into the reactor container When the pressure inside of the reactor goes higher, since the liquid surface in the water vessel is forced down, water is pushed up into the pipeline Since the waterhead pressure of a column of water in the pipeline and the pressure of the reactor container are well-balanced, gases in the reactor container are not leaked to the outside Further, in a case if a great positive pressure is formed in the reactor container, the inner pressure overcomes the waterhead of the column of water, so that the gases containing radioactive aerosol uprise in the pipeline Since water and the gases flow being in contact with each other, this can provide the effect of removing aerosol (T M)

57
Reactor container. Ishiyama, Takenori (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-122300/A/ 9 May 1990 Filed date 1 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

Cooling water channels are disposed at the upper surface of a reactor container in a BWR type reactor Water flowing through the cooling water channels is always supplied from an outer circumferential pool of the reactor container disposed at the outer circumference of the container by a feed water unit Then, the water supplied to the cooling water channels prevails over the entire flow channel, flows along the side wall of the reactor container and returns to the outer circumferential pool of the reactor container Steams filled inside of the reactor container upon accidents are condensated under cooling by cooling water flowing through the cooling water channels at the upper surface of the reactor container and put to natural convection Heat conduction to the cooling water flowing through the cooling water channel can be kept at a high level and the amount of heat removed can be ensured Accordingly, a dry well cooler and a heat exchanger are no more necessary and the reactor container can be cooled stably for a long period of time (I N)

58
Natural convection type BWR reactor. Tobimatsu, Toshimi (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-120693/A/ 8 May 1990 Filed date 31 Oct 1988 4p (In Japanese) JAPIO Also available from INPADOC

In a natural convection type BWR reactor, a mixed stream of steams and water undergo a great flow resistance In particular, pressure loss upon passing from an upper plenum to a stand pipe and pressure loss upon passing through rotational blades are great Then, a steam dryer comprising laminated dome-like perforated plates and a drain pipe for flowing down separated water to a downcomer are disposed above a riser The coolants heated in the reactor core are boiled, uprise in the riser as a gas-liquid two phase flow containing voids, release steams containing droplets from the surface of the gas-liquid two phase, flow into the steam dryer comprising the perforated plates and are separated into a gas and a liquid The dried steams flow to a turbine passing through a main steam pipe and the condensated droplets flow down through the drain pipe and the downcomer to the lower portion of the reactor core In this way the conventional gas liquid separator can be saved without lowering the quality of steam drying to reduce the pressure loss and to improve the operation per formance (N H)

59
Fuel assembly. Kubo, Hiroshi (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-116789/A/ 1 May 1990 Filed date 27 Oct 1988 5p (In Japanese) JAPIO Also available from INPADOC

The present invention concerns a fuel assembly for use in BWR type reactors A plurality of nozzle-like through holes are disposed on the side of a lower tie plate at a position corresponding to the lower end of a channel box Coolants are discharged to the outside of the fuel assembly passing through the through holes Since the shape of the through holes is nozzle-like, the area of the flow channel is decreased to lower the pressure of water by so much as the increase of the flow speed Accordingly, the channel box undergoes a force urging to the lower tie plate due to the difference of water pressure which can suppress the creep deformation of the channel box toward the expanding direction In view of the above the gap between the lower tie plate and the channel box can be kept appropriately to prevent the aging change for the amount of the coolant leakage (I N)

60
Fuel assembly. Yamashita Jun ichi Ueki Taro Ozawa Michihiro (to Hitachi Ltd , Tokyo (Japan)) Japan Patent 2-112795/A/ 25 Apr 1990 Filed date 21 Oct 1988 26p (In Japanese) JAPIO Also available from INPADOC

Control rods are arranged by combining a plurality of square lattice arrangements to increase an area of coolants flow channels at the periphery of a fuel assembly, thereby increasing a thermal margin and a power peaking Change of the void coefficient is increased to reduce the control rods operation by disposing a spectral shift rod having a coolant uprising channel which opens below a fuel support and a coolant descending channel which opens above the fuel support and having a smaller flow channel area than that of the uprising channel Further, the region in a proportional relationship between the differential pressure of entrance/exit and the void coefficient is made greatest by increasing the ratio of the flow area between both of the channels to more than 25 times Further, the side end of the spectral shift rod is directed to the corner of the fuel assembly to enhance the effect and the side walls are cooled efficiently by members and plates After the comple tion of the power operation, the reactor power is controlled by controlling the level of the coolants formed in the spectral shift rod (N H)

61
Automatic process fluid leak detector for research and production power plants. Imperiali, F (to ENEA, Rome (Italy)) Italy Patent Application 48455A89 [1990] Filed date 13 Oct 1989 35p (In Italian) ENEA, Rome (Italy) Uff Brevetti

This paper describes the design of a patented (Italian) automatic leak detector particularly suited for the monitoring of high pressure containers or circuits such as the primary circuit of PWR or BWR type reactor The device, capable of real or almost real time leak detection, consists of a series of three distinct and separate detectors The first analyzes, through a telecamera, visible spectrum signals, the second through a thermograph, analyzes infrared spectrum signals and the third using a chemical excitation mass spectrometer qualitatively and quantitatively analyzes every variation in atmospheric composition An integrated set of data acquisition and processing systems elaborates the data supplied by the sensors, makes comparisons with reference base data and keeps track of the signalling of alarms and the follow up interventions

Light-Water Moderated, Nonboiling Water Cooled

62
(CEA-CONF-10048)
Early detection of deteriorations affecting neutron detectors methods applied in french PWR plants. Bacconnet, E , Burel, J P , Meuwisse, C (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France) Dept d'Electronique et d'Instrumentation Nucleaire) 1989 12p (CONF-8906196-) NTIS (US Sales Only), PC A03/MF A01 Order Number DE91716286

From IAEA IWG-NPPCI specialists' meeting on early failure detection and diagnosis in nuclear power plants' systems and operational experience, Dresden (German Democratic Republic) (20-22 Jun 1989)

The detection of a change in the behaviour of a nuclear instrumentation channel allows the rate of deterioration of the detector, the transmission line and the associated electronic part to be assessed It authorizes preventive replacement of the faulty component Two methods have been developed to improve the availability of neutronic measurement channels of French PWR units 1 - The method referred to as saturation and discrimination curves which is used to check the state of excore neutron detectors, and 2 - The reflectometry method which is particularly suitable for monitoring connection cables and connectors

63
(CNEN-DR-NT-GTT-DAS-05/89)
Input data for simulating Angra I using the TRAC/PF1 code. Madeira, A A , Borges, R C (Comissao Nacional de Energia Nuclear (CNEN), Rio de Janeiro, RJ (Brazil) Dept de Reatores) Dec 1989 128p NTIS (US Sales Only), PC A07/MF A01, OSTI, INIS Order Number DE91607674

This report described the modeling, the input data preparation and the steady state operational conditions results for Angra 1 Nuclear Power Plant obtained with TRAC/PFL Code, aiming the simulation of accidents and operational transients further on Suggestions are presented to minimize the difficulties met in this study (author)

64
(CONF-9010185-2)
Potential impact of enhanced fracture-toughness data on pressurized-thermal-shock analysis. Dickson, T L , Theiss, T J (Oak Ridge National Lab , TN (USA)) [1990] Contract AC05-84OR21400 19p NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91001493

From 18 water reactor safety information meeting, Gaithersburg, MD (USA) (22-24 Oct 1990)

The Heavy Section Steel Technology (HSST) Program is involved with the generation of "enhanced" fracture-initiation toughness and fracture-arrest toughness data of prototypic nuclear reactor vessel steels These two sets of data are enhanced because they have distinguishing characteristics that could potentially impact PWR pressure vessel integrity assessments for the pressurized-thermal shock (PTS) loading condition which is a major plant-life extension issue to be confronted in the 1990's Currently, the HSST Program is planning experiments to verify and quantify, for A533B steel, the distinguishing characteristic of elevated initiation-fracture toughness for shallow flaws which has been observed for other steels Deterministic and probabilistic fracture mechanics analyses were performed to examine the influ ence of the enhanced initiation and arrest fracture toughness data on the cleavage fracture response of a nuclear reactor pressure vessel subjected to PTS loading The results of the analyses indicated that application of the enhanced K_{Ia} data does reduce the conditional probability of failure P(F | E), however it does not appear to have the potential to significantly impact the results of PTS analyses The application of enhanced fracture-initiation-toughness data for shallow flaws also reduces P(F | E), but it does appear to have a potential for significantly affecting the results of PTS analyses The effect of including Type I warm prestress in probabilistic fracture-mechanics analyses is beneficial The benefit is transient dependent and, in some cases, can be quite significant 19 refs , 12 figs , 1 tab

65
(DOE/NE-0086-Rev 1-Background-App)
Department of Energy's team's analyses of Soviet designed VVERs [water-cooled water-moderated atomic energy reactors]. (USDOE Assistant Secretary for Nuclear Energy, Washington DC (USA)) Sep 1989 879p NTIS PC A99/MF A01 OSTI, GPO Dep Order Number DE91002539

This document contains apprendices A through P of this report Topics discussed are a cronyms and technical terms accident analyses reactivity control Soviet safety regulations radionuclide inventory decay heat operations and maintenance, steam supply system concrete and concrete structures seismicity, site information, neutronic parameters, loss of electric power, diesel generator reliability, Soviet codes and standards, and comparisons of PWR and VVER features (FI)

66
(EPRI-NP-6990)
Eddy-current probe characterization. Krzywosz, K J (Electric Power Research Inst , Palo Alto, CA (USA), Jones (J A) Applied Research Co , Charlotte, NC (USA)) Oct 1990 138p Research Reports Center, Box 50490, Palo Alto, CA 94303

Characterization of various eddy current coil configurations was accomplished using state-of-the-art digital testing equipment Program objectives were twofold (1) to determine optimal detection frequencies for different probe diameters and coil configurations with emphasis on the bobbin-coil, and (2) to identify probe types most sensitive to volumetric and planar tube wall degradation Optimal signal response for flaw detection was obtained at a frequency near the probe resonance which is determined by the combined electrical characteristics of the eddy current coil and probe cable length Adding unnecessary lengths of extension cables resulted in poor flaw detection performance The degraded performance was attributed to change in the probe resonance and the associated reduction in the received signal amplitude Probe sensitivity tests were conducted with samples containing volumetric and planar flaws using the measured resonant frequency as a primary detection frequency for each probe type In general, pancake coils offered the highest sensitivity for both volumetric and planar flaws This report summarizes the results of probe characterization and probe sensitivity tests 3 refs , 18 figs , 1 tab

67
(EPRI NP-7008)
Strain-rate damage model for Alloy 600 in primary water. Begley J A (Electric Power Research Inst , Palo Alto CA (USA), Westinghouse Electric Corp , Pittsburgh, PA (USA) Research and Development Center) Oct 1990 101p Research Reports Center, Box 50490, Palo Alto, CA 94303

The objective of this project is to pro vide the mechanical property data required for the development of a strain rate damage model for environmentally-assisted cracking of Alloy 600 tubing and to perform stress corrosion tests from which critical strain rate damage model parameters may be determined for Alloy 600 in primary water environments of PWR reactors An evaluation phase was conducted on sixteen candidate program materials Four heats of material were chosen Tensile tests were performed at several strain rates at temperatures from 75 °F Stress relaxation tests were conducted at 600 °F, 680 °F and 750 °F This material property data can be combined with state-of-the-art finite element creep analyses to compute the strain rate damage function for stress corrosion cracking of Alloy 600 steam generator tubing as a function of geometry and loading Critical strain rate damage parameters were determined for four heats of mill-annealed Alloy 600 tubing in primary water environments at 680 °F by conducting slow strain rate stress corrosion tests as a function of strain rate Effects of prior cold work and variations in primary water chemistry on these critical damage parameters are included 10 refs , 41 figs , 16 tabs

68
(EPRI-NP-7030-D)
Qualification loop tests of cobalt-free hardfacing alloys. Murphy, E V, Inglis, I (Electric Power Research Inst , Palo Alto, CA (USA), Atomic Energy of Canada Ltd , Sheridan Park, ON (Canada) CANDU Operations) Nov 1990 53p Research Reports Center, Box 50490, Palo Alto, CA 94303

AECL is conducting endurance tests on valves hard-faced with four cobalt-free alloys The first phase of the program, in which PWR primary heat transport conditions were simulated in AECL's valve test loop, has reached the 1400 cycle mark This represents approximately 70% of the target value of 2000 cycles The candidate alloys are NOREM 01, NOREM 04, EB 5183 and EVERIT 50 One valve with Stellite 6 trim serves as the standard Prior to loop testing, a baseline inaugural inspection was performed During testing the loop was shut down at approximately 500 cycle intervals, and the valves were disassembled for examination The examinations included seat leak tests, profilometry and nondestructive inspection Corrosion coupons in the loop were used to monitor any material loss due solely to corrosion mechanisms This interim progress report summarizes the examination re sults and the relative performance of the candidate alloys The results to date indicate that, based upon the sliding were damage assessment and seat leakage test results, all the candidate alloys perform better than the Stellite 6 control sample On the same basis, NOREM 04 and EB 5183 are the best of the candidate alloys although there are only minor differences in performance among the four alloys at this time A final assessment can only be made after the cycling tests are complete and the seats and discs have been destructively examined 3 refs 31 figs 9 tabs

69
(ETDE-IT--90 61)
Numerical analysis of a nozzle corner of a 1:5 scale PWR vessel model. Macchi, A , Ponzoni, C , Salani, S , Sampietri, C (Ente Nazionale per l'Energia Elettrica, Milan (Italy) Centro Termica e Nucleare, Centro Informazioni Studi Esperienze (CISE), Milan (Italy)) Feb 1989 11p NTIS (US Sales Only), PC A03/MF A01 Order Number DE91725886

The results obtained through the numerical analysis of a 1 5 scale PWR vessel model are reported in this paper The purpose of the work was the study of the growth of a surface flaw which developed in the nozzle corner region during load cycling Fracture mechanics parameters were evaluated by means of both the finite method and simplified formulae Flaw propagation was studied by Sih s hypothesis as far as it concerns shape change during propagation and by Paris law as far as it concerns crack growth Numerical results were compared with the available experimental ones

70
(EUR--12356)
Human factors reliability Benchmark exercise. Poucet, A (ed) (Commission of the European Communities, Luxembourg (Luxembourg)) Jun 1989 490p NTIS (US Sales Only), PC A21/MF A01

The Joint Research Centre of the European Commission has organized a Human Factors Reliability Benchmark Exercise (HF-RBE) with the aim of assessing the state of the art in human reliability modelling and assessment Fifteen teams from eleven countries, representing industry, utilities, licensing organisations and research institutes, participated in the HF-RBE The HF-RBE was organized around two study cases (1) analysis of routine functional Test and Maintenance (T and M) procedures with the aim of assessing the probability of test induced failures, the probability of failures to remain unrevealed and the potential to initiate transients because of errors performed in the test, (2) analysis of human actions during an operational transient with the aim of assessing the probability that the operators will correctly diagnose the malfunctions and take proper corrective action This report contains the final summary reports produced by the participants in the exercise

71
(EUR--12397)
General and preliminary thermohydraulic, hydrogen and aerosol instrumentation plan for the Phebus Fp-project. Hampel, G , Poss, G Frohlich, H K (Commission of the European Communities, Luxembourg (Luxembourg)) Oct 1989 207p NTIS (US Sales Only), PC A10/MF A01

The objective of the project was to draw up an instrumentation plan for the French core melting programme PHEBUS FP This instrumentation plan essentially was to include proven and reliable instruments for recording various thermohydraulic, aerosol and hydrogen phenomena The candidate measuring methods, which are known mainly from reactor safety programmes, have been described and examined for their usefulness in PHEBUS Each method and instrument has been described in detail under various aspects such as measuring principle, measuring range, technical design, evaluation model, calibration procedure, accuracy, previous experience, commercial availability, etc Special attention has been paid to the behaviour of the measuring transducers when exposed to radiation First, the performance of the instruments was compared with the requirements of PHEBUS The results of this comparison served as the basis for a measuring concept in tabular form, giving the locations of the measurements, the measuring tasks, and the number and kind of instruments that are recommended Redundancy and cost-benefit aspects have been taken into account in qualitative terms

72
(EUR--12398)
Radionuclides measurements for the Phebus FP project - preliminary study. Schuster, E , Nopitsch, K (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 53p NTIS (US Sales Only), PC A04/MF A01

This report gives a review of proposed measurement techniques to measure quantitatively release and transport of fission products during and after fission product release tests The measurement devices provided are for example gamma spectrometers, scintillation counters and sampling vessels The second part reviews the behaviour of fission product iodine under severe accident conditions and describes the iodine sampling system and the determination of iodine species in aqueous solutions

73
(EUR--12399)
Review of analytical techniques to determine the chemical forms of vapours and aerosols released from overheated fuel. Bowsher, B R , Nichols, A L (Commission of the European Communities, Luxembourg (Luxembourg)) Dec 1989 99p NTIS (US Sales Only), PC A05/MF A01

A comprehensive review has been undertaken of appropriate analytical techniques to monitor and measure the chemical effects that occur in large-scale tests designed to study severe reactor accidents Various methods have been developed to determine the chemical forms of the vapours, aerosols and deposits generated during and after such integral experiments Other specific techniques have the long-term potential to provide some of the desired data in greater detail, although considerable efforts are still required to apply these techniques to the study of radioactive debris Such in-situ and post-test methods of analysis have been also assessed in terms of their applicability to the analysis of samples from the Phebus-FP tests The recommended in situ methods of analysis are gamma-ray spectroscopy, potentiometry, mass spectrometry, and Raman/UV-visible absorption spectroscopy Vapour/aerosol and deposition samples should also be obtained at well-defined time intervals during each experiment for subsequent post-test analysis No single technique can provide all the necessary chemical data from these samples, and the most appropriate method of analysis involves a complementary combination of autoradiography, AES, IR, MRS, SEMS/EDS, SIMS/LMIS, XPS and XRD

74
(EUR--12488)
Eliciting and communicating expert judgments: methodology and application to nuclear safety. von

Winterfeldt, D (Commission of the European Communities, Luxembourg (Luxembourg)) 1989 43p NTIS (US Sales Only), PC A03/MF A01

Expert judgment has always been used informally in the analysis of complex engineering problems Increasingly, however, the use of expert judgment has been formalized by eliciting judgments in an explicit, documented and often quantitative way In nuclear safety studies the need for formal elicitation of expert judgments arises because of the lack of data and experiences, the need to adapt model results to the specific circumstances of a plant, and the large uncertainties surrounding the events and quantities that characterize an accident sequence The recognition of the need for a formal elicitation of expert judgments has led to one of the most extensive expert elicitation processes to date in the context of the NUREG 1150 study About 30 safety issues were quantified using expert judgments about probabilities of various uncertain events and quantities, ranging from the failure of a check valve in the cooling system to the pressure built up due to hydrogen production to release fractions of various radionuclides In total, some 1000 probability distributions were elicited from some 50 experts This paper first motivates the use of formal expert elicitation in complex engineering studies and describes the methodology of formal expert elicitation Subsequently, it describes the overall approach of NUREG 1150 and provides an example of the elicitation of the probability of a bypass failure in a pressurized water reactor The paper ends by discussing some lessons learned, problems encountered and by providing some recommendations

75
(EUR–12528)
Analysis of experiments of the University of Hannover with the Cathare code on fluid dynamic effects in the fuel element top nozzle area during refilling and reflooding. Bestion, D (Commission of the European Communities Luxembourg (Luxembourg)) Nov 1989 23p NTIS (US Sales Only), PC A03/MF A01

The CATHARE code is used to calculate the experiment of the University of Hannover concerning the flooding limit at the fuel element top nozzle area Some qualitative and quantitativ limit at the fuel element top nozzle area on both the actual fluid dynamics which is observed in the experiments and on the corresponding code behaviour Shortcomings of the present models are clearly identified New developments are proposed which should extend the code capabilities

76
(EUR–12578, pp 21-35)
Operating experience and TPA: the Italian perspective. Grimaldi, G (ENEA Rome - (IT)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

Collection and analysis of operating experience from the Italian plants and utilization of abroad data both to plants in operation and in construction are presented Some results are also referred, aimed to evidence the role of the international cooperation to safe operation of nuclear plants The approach to the Trend and Pattern analyses is described as well, and the use of computerized techniques of analysis on personal computer Finally on going activities are introduced, specifically application of operating experience of plants in operation to small sized reactors and to ones with more intrinsic safety characteristics, review of the reporting system for future application and comparative analysis of the different realization of selected safety systems

77
(EUR–12578, pp 185-198)
Incident analysis, data gathering and use of statistics for operational purposes. Girault, B (Electricite de France (EDF), 75 - Paris (France)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The Nuclear and Fossil Generation Division of Electricite de France has developed a database for operational purposes Operational means that the initial analyses and the direction taken adopted at later stages are essentially directed towards experience feedback Consequently, requirements of precision, coherence and efficiency characterize the causal analysis applicable to numerous events, by numerous users, over a long period This use of many analysts, using common methods over a long period of time assures the quality of the final results of the data base The use of the results is illustrated in a study of safety-related incidents The study resulted in a number of specific remedies that were applied in the French power plants

78
(EUR–12578, pp 351-362)
Presentation of a method for the sequential analysis of incidents. Delage, M , Giroux, C , Quentin, P (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (FR) Inst de Protection et de Surete Nucleaire) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

This paper presents a method which is designed to assist in the analysis of safety and is based on the graphic representation of the occurrence of incidents significant for safety in 900 MWe PWR units The graphs obtained are linked together to produce a general tree of events With this tool, and on the basis of operating experience, we are then able to imagine complex incident scenarios, to evaluate the potential consequences of a particular incident or to seek out the causes which could lead to a given event Interactions between systems or common mode faults can also appear with this method

79
(EUR–12578, pp 379-393)
Statistical prediction of the numbers of degraded tubes in nuclear power plant steam generators. Gallucci, R H V , Klisiewicz, J W , Craig, K R (Combustion Engineering, Inc Windsor, CT (US)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

Corrosion of nuclear power plant steam generator (SG) tubes often necessitates plugging/sleeving, causing decreased SG thermal performance and possible SG replacement Statistical methods have been developed to predict probabilistically the numbers of tubes degraded due to secondary side pitting, wastage, and intergranular attack/stress-corrosion cracking Inspection data from two Combustion Engineering (C-E) plants have been converted into statistics representing defect formation and growth Computer simulation programs have been generated to predict the numbers of tubes to be plugged/sleeved during future outages The probabilistic predictions for both plants successfully have bounded subsequent observations While so far applied only to C E SGs for the three degradation phenomena, the statistical methodology is adaptable to other SG types and phenomena

80

(EUR–12578, pp 407-418)

Trend analysis for incidents of steam generator tube rupture and loss of electric power based on IRS reports. Fujiki, K , Ishigami, T , Namatame, K (Japan Atomic Energy Research Inst Tokai Ibaraki, (JP)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

It is widely recognized the importance of the feedback of operational experiences to design, construction, and operation of the nuclear power plant For this purpose, it is essential to exchange incident informations internationally, because they frequently include common features which can be applied to nuclear power plants in each member country JAERI has already established database management and retrieval system for NEA/IRS informations and been carrying out analysis of IRS reports in order to provide useful informations for regulatory bodies Present paper describes the method adopted in recent analysis of specific type of incident i e steam generator tube leakage incidents including tube ruptures in pressurized water reactors and loss of electric power incidents in boiling water reactors The results of analysis are also shown

81

(EUR–12578, pp 437-448)

Reliability of emergency diesel-generators used in french NPP evaluation of the failure rate and its trend failures and dysfunctions review. Colas, A F (CEA Centre d'Etudes Nucleaires de Fontenay-aux Roses 92 (FR) Dept d Analyse de Surete) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

Emergency Diesel-Generators (EDG) reliability evaluation is based on examination of tests and operation abnormalities collected in national computerised data bank We gather all available data in order to establish failures rate annual values and to follow their trend Technical analysis aims at identifying failures modes in order to find palliative or curative solutions The present paper tries to show our main findings and the way of technical approach we follow in this matters

82

(EUR–12578, pp 575-583)

Instrument validation system of general application. Filshtein, E L (Combustion Engineering, Inc Windsor, CT (US)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

This paper describes the Instrument Validation System (IVS) as a software system which has the capability of evaluating the performance of a set of functionally related instrument channels to identify failed instruments and to quantify instrument drift Under funding from Combustion Engineering (C-E), the IVS has been developed to the extent that a computer program exists whose use has been demonstrated The initial development work shows promise for success and for wide application, not only to power plants, but also to industrial manufacturing and process control Applications in the aerospace and military sector are also likely

83

(EUR–12578, pp 633-641)

Constitution of an incident database suited to statistical analysis and examples. Verpeaux, J L (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (FR) Dept d'Analyse de Surete) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

The Nuclear Protection and Safety Institute (IPSN) has set up and is developing an incidents database, which is used for the management and analysis of incidents encountered in French PWR plants IPSN has already carried out several incidents or safety important events statistical analysis, and is improving its database on the basis of the experience it gained from this various studies A description of the analysis method and of the developed database is presented

84

(LA–11850-MS)

Underwater measurement of a 15 × 15 MOX PWR [mixed oxide pressurized water reactor]-type fuel assembly. Nelson, A J , Bosler, G E , Augustson, R H , Cowder, L R (Los Alamos National Lab , NM (USA)) Dec 1990 Contract W-7405-ENG-36 71p (ISPO–316) NTIS, PC A04/MF A01 - OSTI, GPO Dep Order Number DE91002950

Underwater measurements have been made on a fresh, mixed oxide, 15 × 15 fuel assembly to evaluate various nondestructive assay techniques for determining the plutonium content in the assembly For these laboratory measurements, best results were obtained with assays using multiplication-corrected neutron coincidence count rates

85
(NUREG/CR-5524-Vol 2)
TMI-2 Vessel Investigation Project (VIP) Metallurgical Program. Diercks, D R (Nuclear Regulatory Commission, Washington, DC (USA) Div of Engineering, Argonne National Lab, IL (USA)) Nov 1990 Contract W-31109-ENG-38 65p (ANL-90/34-Vol 2) NTIS, PC A04/MF A01 - GPO, OSTI, INIS

This report summarizes the work performed by Argonne National Laboratory (ANL) on the Three Mile Island Unit 2 (TMI-2) Vessel Investigation Project (VIP) Metallurgical Program during the nine month period from October 1989 through June 1990 During the reporting period, a series of heat treatment experiments on archive material from the lower head of the Midland nuclear reactor has been completed, the resulting microstructures have been examined, and hardness values have been determined The results have been compared with the predictions of published continuous-cooling transformation diagrams for A533 Grade B steel Round-robin microstructural characterizations and mechanical tests on the archive material have also been completed by the participating Organization for Economic Co-operation and Development (OECD) partner laboratories Decontamination of samples from the TMI-2 lower head is underway at ANL, the procedure being utilized is described in this report Detailed microstructural and scanning electron microscope examinations of Specimen E-6 have been carried out in an attempt to determine the extent and cause of cracking observed in the clad surface The results of microstructural observations, hardness measurements, and tensile tests indicate that the base metal in the vicinity of the crack attained a maximum temperature of $\approx$1000 to 1100°C during the accident The molten fuel apparently did not penetrate into the cracks and interact with the base metal It was tentatively concluded that the cracking of the cladding in Sample E-6 probably occurred during the early stages of cool-down after the accident, when the still hot cladding layer was placed into tension because of thermal contraction 7 refs, 52 figs, 17 tabs

86
(NUREG/CR-5617)
Auxiliary feedwater system risk-based inspection guide for the J.M. Farley Nuclear Power Plant. Vo, T V, Pugh, R, Gore, B F, Harrison, D G (Nuclear Regulatory Commission, Washington, DC (USA) Div of Radiation Protection and Emergency Preparedness, Pacific Northwest Lab, Richland, WA (USA)) Oct 1990 Contract AC06-76RL01830 26p (PNL-7349) NTIS, PC A03/MF A01 - GPO, OSTI, INIS

In a study sponsored by the US Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory has developed and applied a methodology for deriving plant-specific risk-based inspection guidance for the auxiliary feedwater (AFW) system at pressurized water reactors that have not undergone probabilistic risk assessment(PRA) This methodology uses existing PRA results and plant operating experience information Existing PRA-based inspection guidance recently developed for the NRC for various plants was used to identify generic component failure modes This information was then combined with plant-specific and industry-wide component information and failure data to identify failure modes and failure mechanisms for the AFW system at the selected plants J M Farley was selected as the second plant for study The product of this effort is a prioritized listing of AFW failures which have occurred at the plant and at other PWRs This listing is intended for use by NRC inspectors in the preparation of inspection plans addressing AFW risk-important at the J M Farley plant 23 refs, 1 fig, 1 tab

87
(PSI-78)
Sensitivity analysis of the rod ejection accident for the Beznau reactor. Saphier, D, Zimmermann, M A, Knoglinger E, Jacquemoud, P (Paul Scherrer Inst (PSI), Villigen (Switzerland)) Aug 1990 132p NTIS (US Sales Only) PC A07/MF A01, OSTI, INIS Order Number DE91610159

The rod ejection accident (REA) of the Beznau (KKB-2) nuclear power plant was investigated The REA analysis was performed using the RETRAN-02 computer code Four basic cases were investigated for the cycle 16 conditions At the beginning-of-life (BOL) the hot full power (HFP) and the hot zero power (HZP) were simulated At the end-of-life (EOL) conditions, again the HFP and HZP cases were simulated The RETRAN-02 code was modified to permit the inclusion of a time dependent power distribution function which in a REA accident changes significantly from the normal power distribution This change in the reactor power distribution shape takes place during a very short period of time Further modifications to the code permitted the Doppler and void reactivity coefficients to be spatially modified as a function of time The report discusses in detail the parameters of importance in the REA analysis and the degree of conservatism that should be included in the calculation The methodology of the present study is outlined in detail and the result of the four basic REA cases are presented A detailed sensitivity study was carried out to investigate the sensitivity of all the parameters that can affect the core behavior during REA The sensitivity of the calculated results to each of the input parameters is also presented in detail From the present study it can be concluded that a single REA does not have significant radiological consequendes affecting the Beznau reactor environment (author) 113 figs, 24 refs

88
(ZJE-282)
Experience with the WWER-440 MW reactor pressure vessel in-service inspections and evaluation of their results. Brumovsky, M, Kralovec, J, Prepechal, J, Sulc J (Skoda, Plzen (Czechoslovakia) Zavod Vystavba Jadernych Elektraren) 1989 11p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91610160

The Power Machinery Plant of Skoda Works in Plzen carries out in-service inspections of WWER-440 MW reactor pressure vessels by means of remote controlled inspection equipment - the TRC reactor test system, and some other inspections devices The results of the in-service inspections were evaluated by methods based on the fracture mechanics approach, the knowledge of stress and strain distribution, and the operating history of the pressure vessels Examples of types of defects found and their analysis are shown (author) 1 tab

89
The Physics of Reactors: Operation, Design and Computation.

Volume 1. Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

These proceedings of the International Conference on the physics of reactors Operation, Design and Computation, Volume 1, are divided into 7 oral sessions bearing on - High conversion light water reactors and actinide burners 8 papers - Sodium cooled fast reactor design 7 papers - Nuclear data measurement and evaluation 6 papers - Nuclear data delayed neutrons 5 papers - Integral experiments thermal reactors 6 papers - Integral experiments fast reactors 6 papers - Use of MOX fuel in thermal reactors 10 papers - Core physics parameters validation for LMFBR 6 papers

90

Rules for qualification of KRT health physics channels in EDF power plants. Mallet, Y , Houin, J M (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp 747-754 of Operability of nuclear Systems in normal and adverse environments Volume 2 Paris (FR), Societe Francaise d'Energie Nucleaire (1989) 422p (CONF-890911-)

From International conference on operability of nuclear systems in normal and adverse environments, Lyon (France) (18-22 Sep 1989)

The principle for qualification of health physics channels results from a safety classification of the latter, which is applied from the outset The recent development of a channel for measuring primary/secondary leakage rate led to qualification of VAMCIS channel based on the same principles and using classification 1E (qualification K3) A presentation of the qualification programme and test results will illustrate the general qualification principles

91

The convertible spectral shift reactor. Millot, J P (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp I 1 I 33 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

In view of the surplus plutonium stocks that will be available in France by the beginning of the 21st century (on the order of 100 tonnes), and taking into account an increasing rate of consumption of the quantity produced annually by the spent-fuel reprocessing plants (approximately 6 5 tonnes), Framatome has developed the concept of an optimized plutonium-using light water reactor This is the convertible spectral shift reactor, known in France as the RCVS (for Reacteur Convertible a Variation de Spectre)

92

Preliminary design and analysis of a slightly-enriched spectral shift reactor. Grove, R E , Martin, W R , Lee, J C , Oukebdane, A , Keller, P M , Edlund, M C (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp I 34-I 44 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The slightly-enriched spectral shift reactor (SESSR) utilizes conventional light water reactor fuel arranged in a moderately-tight lattice with spectral shift control rods (SSCRs) which displace water when inserted The SSCRs are inserted in the beginning of the cycle, hardening the spectrum and increasing the production of fissile plutonium The SSCRs are withdrawn later in the cycle, softening the spectrum and depleting the plutonium Recycling of plutonium is not necessary, avoiding difficulties such as a tendency for a positive moderator coefficient and regulatory concerns associated with plutonium recycle in the United States Preliminary fuel cycle calculations including equilibrium core analyses, have shown that the use of spectral shift control can result in 15-22% increases in fuel cycle length This spectral shift advantage tends to decrease with tighter lattices, leading to the conclusion a hexagonal lattice may not be necessary to achieve satisfactory fuel cycle performance

93

Venus-2 PWR engineering mock-up: core qualification and fast neutron field characterization. Ait Abderrahim, H , D'hondt, P , Fabry, A , Leenders, L , Minsart, G (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp IV 23-IV 31 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

VENUS-2 is the second stage of the VENUS PWR engineering Mock-up experiment which is part of the Belgian PWR-Pressure Vessel Surveillance Programme, sponsored by the Belgian utilities Through a cooperation agreement it contributes also, as one of the benchmark fields, to the international LWR-Pressure Vessel-Surveillance Dosimetry Improvement Program supported by the U S Nuclear Regulatory Commission The VENUS-2 Pressure Vessel Mock-up simulates a low-leakage core with the reflector geometry and the core boundary shape of a generic 3-loop power plant The low leakage configuration is achieved by replacing the eight outer rows of the stainless steel cladded, 4 w/o U-235 enriched, UO_2 fuel pins by mixed oxide fuel pins having the same diameter and the same cladding but containing 2 7 w/o Pu and 2 w/o U-235 enriched fuel The comparison of the experimental and the theoretical pin-to-pin power distributions is given in this paper as well as the comparison of computed and measured ex-core fast neutron responses of several threshold detectors The paper gives also some information on the VENUS-3 programme which is designed to benchmark the PLSA (Partial Length Shielded Assembly) concept This concept aimed to lower exposure to critical vessel sections, especially at the horizontal weld level

94

Studies of VVER physics on LR-O experimental reactor. Osmera, B , Holman, M , Rypar, V , Hadek, J , Broulik, J (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp IV 47-IV 56 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The operation of nuclear power plants with VVER-440 reactors, the construction and operation of power plants with VVER-1000 reactors as well as the manufacture of nuclear power installation components put questions to the research in the field of neutron physics the solution of which increases considerably nuclear safety and economy The paper reviews the research methodology of urgent problems such as the increase in the capacity of spent nuclear fuel storages, the determination of pressure vessel radiation loading with higher accuracy, the description of time and spatial transitional phenomena of neutron field with more precision in the core of the reactor and the main results to be applied in VVER-440 operation and VVER-1000 construction and operation A significant part of the experimental programme has being performed under contract and in cooperation with IAE Moscow and Skoda Concern and ZfK Rossendorf near Dresden (DDR)

95
Synthesis of the experimental validation of global fission-product capture for PWR's and advanced reactor lattices. Chaucheprat, P , Mondot, J , Garzenne, C (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp IV 32-IV 46 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation Paris (France) (23-27 Apr 1990)

During the last few years a large effort has been made in France to get an experimental validation of fission-product cross-sections in several spectral conditions Three experiments, the so-called MORGANE/S, MORGANE/R and MELODIE, using the oscillator technique, were performed in the MINERVE facility in CADARACHE, the fission product reactivity worth of irradiated fuel samples for a set of burn-ups (20-60 GWd/t) has been measured in PWR and HCLWR neutron spectra The analysis of these experiments has been carried out using the APOLLO code and two different cross-section libraries (the old Apollo 79 library and the new set of JEF1 cross-sections) The results emphasize very clearly some trends related to the relative importance of the thermal and epithermal energy ranges

96
Advanced strategies for Fragema fuel managements. Guyot, F , Bonniaud, M (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 11-VI 22 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The recent fuel management developments made by FRAGEMA aim to meet plant utility concerns by offering enhanced utilization of fissile material Optimization of fuel reloads can be achieved in different and mutually complementary ways These are reduction of the number of feed assemblies, use of hybrid or IN-OUT loading patterns and/or plutonium recycling These advanced fuel management strategies have been made possible by the wide-ranging refueling experience built up by FRAGEMA and by the high performance level of its fuel, particularly its proven capacity to sustain high burnups

97
French PWR operation feed-back comparison between predicted and measured core physics parameters. Barral, J C , Hervouet, C , Lam-Hime M , Bergeot, M A , Larderet, P , Lefebvre J C , Vassallo, A (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 34-VI 43 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-27 Apr 1990)

Electricite de France operates a large number of 900 MW-class and 1300 MW-class units and has developed its own standard tool for core physics parameters calculation A very significant statistical analysis of PWR core parameter calculation accuracy is possible In spite of high constrained operating conditions (load follow, fuel cycle length flexibility), various core management strategies (three or four uranium batches, uranium enrichment, plutonium recycling) and reloads of different suppliers and designs, results are presented which show a good agreement between predictions and measurements

98
Investigation of high-burnup LWR uranium, MOX and recycled-uranium fuel on the basis of Jef-1 data validated at KFK. Wiese, H W (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 44-VI 52 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Based on the Joint Evaluated File JEF-1, with the KfK burnup code system KARBUS and subsequent KORIGEN calculations the characteristics of high-burnup PWR uranium, mixed-oxide (MOX) and recycled uranium (RU) fuels are analysed Actinide masses, decay heat, radioactivities and radiation are discussed for exposures from 40 - 60 GWd/tHM both in case of reprocessing and in case of a direct storage of used fuel A comparison with recent results from LANL for uranium fuel burnt to 50 GWd/tHM shows good agreement for important actinides and fission products

99
Epicure: an experimental programme devoted to the validation of the calculational schemes for plutonium recycling in PWRs. Mondot, J , Gauthier, J C , Chauchеprat, P , Chauvin, J P , Garzenne, C , Lefebvre J C , Vallee, A (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 53-VI 64 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

In France, three 900 MWe PWR cores are currently loaded with UO_2 and MOX subassemblies (namely St Laurent-B1 and B2, and Gravelines-4) Due to the difficulties which arise in such new situations to assess uncertainties on the main parameters (related to safety, operation load following etc) operational safety margins have been increased in particular those concerning the radial peaking factor

The EPICURE programme was ageed in 1987 within the framework of a collaboration between CEA, EdF and FRAMATOME This experiment, which is planned to be performed from 1989 to 1991 in the EOLE critical facility at the Nuclear Center of Cadarache, is aimed at improving the validation of the calculational schemes of PWRs reactors containing MOX subassemblies In this paper we present the general features and the objectives of the programme together with a first analysis of the results presently available Calculations using the APOLLO cell code, the BISTRO XY Sn transport code and the 99 groups CEA/89 library are compared to experimental results concerning the neutronic balance of the UO_2 reference core (fundamental mode and 2 dimensional power distribution) and the pin by pin fission rate measurements in the MOX UO_2 core simulating a first MOX reload Finally a first validation of the power distribution derived from in core fission chamber measurements is also presented

100

100% plutonium recycling feasibility in a 900 MWe PWR core. Bergeron, J , Lenain, R , Loubiere, S (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 1-VI 10 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

This study is focalized on a control mean research for a 900 MWe PWR loaded only with mixed oxyde assemblies The reactivity balances induce temperature limits not to go under for various control modes (black and grey) This study stage leads to the imperative need to conceive very efficient rods Only B_4C seems to satisfy these control conditions So different absorber rod geometries and compositions are examined more precisely The materials considered are on one hand for the rod core, the usual AIC, Hafnium and enriched high density B_4C on the other hand for the clad, SS304 steel and Hafnium

101

Mathematical modelling of dynamic deformations in a cylindrical shell excited by turbulent flow. Kuzelka, V *Nuclear Engineering and Design (Netherlands),* 121 No 1, 39-43 (Jul 1990)

In this paper the mathematical model of the vibration response of structures to random exciting forces is applied to explore the influence of fluid flow parameters on the dynamic deformations and behaviour of a cylindrical shell A physical model of the PWR core barrel, considered as a cylindrical shell supported at both ends, is used to demonstrate by numerical experiments the mentioned influence The mean velocity vector of the fluid flow is the basis for the aero hydrodynamic excitation expressed by a coherence function of fluctuating surface pressure both in the axial and circumferential directions stated as the dependence on the values of the correlation lengths The results of the courses of the generalized spectral loadings of the ms amplitudes of the displacement and stress distributions are shown at the dependence on the flow parameters (orig)

102

Dealing with vibration wear in French pressurized water reactor internals. Cauquelin, C *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 28-30 (Jul 1990)

Wear has been encountered in bottom mounted instrumentation thimbles and, more recently, in control rods at French Pressurized Water Reactors Both problems, which arise from flow induced vibration, have been extensively modelled and analysed Effective solutions have been developed and implemented (author)

103

Helping the instructor to instruct. Strong J S *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432 46-48 (Jul 1990)

Westinghouse s Reactor Simulator Instructor System has been ergonomically designed to cut down the number of control manipulations required to carry out functions Full use is made of windowing, mouse control and touchscreen techniques, and two way remote control unit can be provided for the instructor Due to its modular nature and well-defined ASCII mnemonic command structure, the system can also be installed on non-Westinghouse simulators (author)

104

Developing and upgrading full-scope systems in Czechoslovakia. Krcek, V , Leicman, J *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 48-49 (Jul 1990)

Czechoslavakia has one full-scope VVER type reactor simulator in operation and two more are planned, making maximum use of indigenous computer technology The existing facility has recently been upgraded to give a truer representation of plant functions (author)

105

TREAT Nuclear Plant Analyzer aids engineering and training. Desaedeleer, G , Paternoster, Y Hancart V *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 50 51 (Jul 1990)

Westinghouse s TREAT (Transient Real-time Engineering Analysis Tool) Nuclear Plant Analyzer combines a multi purpose thermohydraulic code with a user friendly, modular, real-time simulation environment The system provides a powerful desk-top simulation and analysis tool with high resolution graphics for a fairly moderate investment (author)

106

Kansai takes a predictive approach to degradation problems. Mori, M , Matmura, H *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 40, 42 (Jul 1990)

As operating experience has accumulated at Kansai Electric Power's Pressurized Water Reactor plants, various types of component degradation phenomena have presented the utility with unexpected challenges To address this kind of problem Kansai has developed extensive capabilities in the area of preventive maintenance which are described here (author)

107

Radiation embrittlement and the annealing of Soviet PWR vessels. Weeks, J R *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 33-36 (Jul 1990)

The US Department of Energy has been carrying out in-depth evaluations of Soviet PWR technology, including assessments of the problem of radiation embrittlement and the efficacy of the measures proposed by the Soviets, which include annealing This article reviews the findings (author)

108
Paks surveillance shows reactor vessels safe from embrittlement. Trampus, P *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 36-37, 40 (Jul 1990)

Surveillance of Hungary's Paks VVER 440 Pressurized Water Reactors by means of "Loviisa-type" shows that embrittlement of the reactor pressure vessels should not be a problem during their operational lifetime Each reactor pressure vessel has six sets of specimens consisting of base metal (Cr-Mo-V alloyed steel), heat affected zone and weld A set consists of twelve Charpy V specimens, twelve Charpy size 10 three-point band specimens and six tensile specimens Two sets are for studying thermal embrittlement The specimens are located between the core and the vessel walls (author)

109
Seeing beyond the design basis with the STRATEG glass model. *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 44-46 (Jul 1990)

Rhineland-Westphalia Power Stations AG has built a 1 10 scale model of a Pressurized Water Reactor from glass, enabling trainees to see exactly what is going on inside the system Practically all the German utilities have taken advantage of this system, and trainees are much better prepared for their subsequent work on full-scale simulators (author)

110
Using eddy currents to examine PWR control rod wear. Dobbeni, D *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 48-50 (May 1990)

Laborelec of Belgium has pioneered the application of eddy currents to examining wear in Pressurized Water Reactor rod cluster control assemblies - a new phenomenon recently encountered at several plants Careful design and extensive laboratory simulation was needed to cope with underwater conditions the presence of boron and highly irradiated materials The new system provides a fast and efficient way to monitor rodlet integrity without lengthening the outage period (author)

111
Managing large amounts of erosion - corrosion NDE data with CEMS. Bogard, T , Batt, T , Roarty, D , Hruby, R *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 50-54 (May 1990)

Since the nuclear industry is dramatically increasing the amount of non-destructive examination for erosion-corrosion, managing the large quantity of data arising from these examinations has become a key issue Westinghouse's PC-based Corrosion-Erosion Monitoring System (CEMS) takes wall thickness data from standard nondestructive examination devices and creates tables and 2D and 3D plots of worn components, makes component life predictions, and evaluates structural integrity Reports are generated automatically (author)

112
Increasing power painlessly at Doel 3 and Tihange 2. Hillegeer, M , Bruyere, M *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 58-59 (May 1990)

Feasibility studies have shown that it is possible to both increase the power level and lengthen the operating cycle at the Belgian Doel 3 and Tihange 2 Pressurized Water Reactors, with few modifications to the equipment or protection system setpoints (author)

113
Using FARIS [Fuel Assembly Repair and Inspection Station] for assembly clean-up and debris removal. Tucker, J S , Sapyta, J J *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 60-62 (May 1990)

Because fuel inspection and repair tasks are commonly done on the critical path during plant refuelling outages, they must be completed quickly and efficiently with minimal costs To fulfil these demands, the Babcock and Wilcox Fuel Company has designed a Fuel Assembly Repair and Inspection Station (FARIS) for fuel assembly clean-up and debris removal in Pressurized Water Reactors The system is portable and can also be used for carrying out visual inspections on fuel assemblies, spacer grid repair, fuel rod oxide thickness measurements and for fuel rod water channel inspections (author)

114
Developing ultrasonics for PWR pump bowl in-service inspection. Dombret, P , Caussin, P , Rorive, P *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 42-44 (May 1990)

An effective solution to the problem of inspecting thick-section austenitic stainless steel in Pressurized Water Reactor pump bowls has been developed in Belgium, using ultrasonic beam focusing The first application was at Tihange 1 using dedicated automated equipment Wider applications are envisaged (author)

115
Nuclear Electric gains new insights into the problem of cast austenitics. Gilroy, K S *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430, 44-46 (May 1990)

Nuclear Electric's work to assure the integrity of the Sizewell B Pressurized Water Reactor pump bowls has furthered understanding of how ultrasound propagates in austenitics It has demonstrated that even in coarse-grained austenitic castings it is possible to devise inspection techniques that can discriminate between insignificant flaws and actual cracks, and that experienced, carefully trained operators can carry out such inspections with few false calls (author)

116
Study and modelling of recirculating and stratified Turbulent flows: Industrial point of view. Baron, F *Bulletin de la Direction des Etudes et Recherches, Electricite de France, Serie A Nucleaire, Hydraulique, Thermique (France),* No 1, 1-29 (Mar 1990)

Incompressible turbulent flows are a basis for implementation and improvement of general-purpose turbulence modelling Emphasis is laid on one-point turbulence closures (including second-moment ones) and on a typical industrial field of study and application, namely water circulation in pressurized water nuclear reactors (PWR)

117
Thermohydraulic and mechanical behaviour of nuclear unit moisture separator reheater. Eddi, M , Stephan, J M *Bulletin de la Direction des Etudes et Recherches, Electricite de France, Serie A Nucleaire, Hydraulique, Thermique (France),* No 1, 45-59 (Mar 1990) (In French)

The reheater of the Moisture Separator-Reheaters is a complex heat-exchanger because - it condenses steam on the inside of the tubes - it suffers high discard of temperature on transient situations, - it is formed with great different inertia steels These particularities have justified this double study of the thermohydraulic and thermomechanical behaviour of this kind of heat exchanger Thermohydraulic simulations with the SICLE computer code in severe transient conditions such as *load rejection reveal hard condensation* fronts and flow reversals in the lower branches of the reheater For modelling these condensation fronts, modifications made to the SICLE numerical model in combination with a very fine schematization of the reheater have been necessary The comparison between calculated and on site measured temperatures is highly satisfactory The analysis of the mechanical calculus reveals the most important parameters influencing the reheater mechanical behaviour Beside the considerable difference of the thermal expansion coefficient values of the *steels used for the tubes and the* casing, we must notice high friction coefficient value and the importance of the manufacturing tolerances

118
Results of 11th regular inspection of No.1 plant in Genkai Nuclear Power Station, Kyushu Electric Power Co., Inc. (Agency of Natural Resources and Energy, Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 25-26 (Mar 1990) (In Japanese)

The 11th regular inspection of No 1 plant in Genkai Nuclear Power Station was carried out from March 6 to October 5 1989 The parallel operation was resumed on September 7, 1989, 186 days after the parallel off The facilities which were the object of inspection were the reactor proper, reactor cooling system, measurement and control system, fuel facilities, radiation control facilities, waste facilities, reactor containment installation, and emergency power generation system On these facilities which were the object of inspection, the appearance, disassembling, leak function, performance and other inspections were carried out As the result, significant signals were observed in 410 steam generator heating tubes, but other abnormality was not found The works related to this regular inspection were accomplished within the range of the limit of dose equivalent based on the relevant laws The main reconstruction works carried out during the period of this regular inspection were the repair of defective steam generator heating tubes and the replacement of a LP turbine disk (K I)

119
Results of 10th regular inspection of No.1 plant in Mihama Power Station, Kansai Electric Power Co., Inc. (Agency of Natural Resources and Energy Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 26-27 (Mar 1990) (In Japanese)

The 10th regular inspection of No 1 plant in Mihama Power Station was carried out from March 24 to August 30 1989 The parallel operation was resumed on August 10, 1989, 140 days after the parallel off The facilities which were the object of inspection were the reactor proper, reactor cooling system, measurement and control system, fuel facilities, waste facilities, radiation control facilities, reactor containment installation and emergency power generation system On these facilities which were the object of inspection, the appearance, disassembling, leak, function, performance and other inspections were carried out As the result, significant signals were observed in 6 steam generator heating tubes, but *other abnormality was not found* The works related to this regular inspection were accomplished within the range of the limit of dose equivalent based on the relevant laws The main reconstruction works carried out during the period of this regular inspection were the replacement of two diesel generators, the repair of in core structures in the reactor vessel, the repair of thermometer bypass piping for primary coolant, and the repair of defective steam generator heating tubes (K I)

120
Studying separation and hydrodynamic properties of a steam generator for the commercial power unit with the WWER-1000 reactor. Ageev, A G , Korol'kov, B M , Dants, V G , Ipatov, P L , Kontsevoj, A A , Vasil'eva, R V , Nekrasov, A V , Pikus, V Yu , Titov, V F , Tarankov, G A *Ehlektricheskie Stantsii (USSR),* No 1, 29-33 (Jan 1990) (In Russian)

Possible ways of improving design of the PGV-1000 steam generators (SG) are investigated It is shown that installation of additional submerged perforated sheets (SPS) under the gap between SG body and SPS bead from the side of hot collector enables to remove completely the ejection of steam-water mixture from the gap This results in sufficient decrease of humidity at separator input, improvement of steam separation conditions and provides the decrease of humidity level in steam line Installation of additional SPS doesn't change hydrodynamics of water volume in the steam generator

121
Balancing of highly flexible shaft lines on their critical bending speeds. Chevalier, R *RFM (Revue Francaise de Mecanique) (France),* No 1 17-22 (1990) (In French)

The balancing of EDF shaft lines has been performed for a decade with the help of the multiplane balancing method, using coefficients of influence *at nominal speed The method makes* it possible to seek the minimum level of vibrations with the smallest possible corrective weights, using the least squares pseudo-inverse optimization technique Due to the flexibility of the large shaft lines placed into service in the last few years, it is necessary to balance not only at nominal speed but also at critical bending speeds Accordingly, we have developed a new method which combines the efficiency of modal balancing with the simplicity of balancing with *coefficients of influ*ence and which finds an optimum balancing for nominal and critical speeds thanks to its weightings option (for machines with low modal damping) The data analysis and balancing programs can run on of desk computers such as the HP 200, 300 and 500 series and allow the corrective weights to be determined immediately, on-site, from the data provided by EDF line shaft monitoring systems

122
PIUS - the nuclear reactor of tomorrow. Nuclear power plants with 'passive' safety. Hannerz, K , Pedersen, T *ABB Review (Switzerland),* No 2, 3-14 (1990)

German version published in ABB Tech (1990) (no 2) p 3-14

The core of the PIUS (Process Inherent Ultimate Safety) *reactor* is protected by a unique reactor system configuration and a special thermohydraulic feedback mechanism As this guarantees the core integrity, the primary task of the surveillance and control systems is to keep the reactor operating In addition to providing unique 'inherent' nuclear safety, this has made a greatly simplified and user-friedly plant possible (orig)

123
Automatic acoustic and vibration monitoring system for nuclear power plants. Tothmatyas, Istvan, Illenyi, Andras, Kiss, Jozsef, Komaromi, Tibor, Nagy, Istvan, Olchvary, Geza *Mueszeruegyi es Merestechnikai Koezlemenyek (Hungary),* 26 No 48, 17-21 (1990) (In Hungarian)

A diagnostic system for nuclear power plant monitoring is described Acoustic and vibration diagnostics can be applied to monitor various reactor components and auxiliary equipment including primary circuit machinery, leak detection integrity of reactor vessel loose parts monitoring A noise diagnostic system has been developed for the Paks Nuclear Power Plant, to supervise the vibration state of primary circuit machinery An automatic data acquisition and processing system is described for digitalizing and analysing diagnostic signals (R P) 3 figs

124
Use of nuclear power plants in development of district heating. Machacek, I , Skalicky, J *Energetika (Prague) (Czechoslovakia),* 40 No 1, 19-22 (1990) (In Czech)

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

Nuclear power plants are envisaged to be the main heat source in Czechoslovakia after the year 2000 Experience with district heating systems supplying heat to distances of up to 59 km is described New problems appear - technical, organizational and technological due to increased pipeline dimensions and length The first Czechoslovak district system supplying heat from the Jaslovske Bohunice nuclear power plant to Trnava was designed in 1982 and finished in 1988 It brought about many organizational problems but its ecological results are good Two more heating systems from Dukovany to Brno and from Blahutovice to Ostrava are planned The operation will be environment-friendly and economical as it will result in savings of conventional fuel (J J) 3 figs

125
Manufacture of the 300 MW steam generator and pressure stabilizer for Qinshan Nuclear Power Station. Qian Yi, Miao Deming *Dongli Gongcheng (Power Engineering (Shanghai)) (China),* 9 No 4, 51-58 (Aug 1989) (In Chinese)

A brief description of the manufacturing process of the steam generator and pressure stabilizer for 300 MWe Qinshan Nuclear Power Station in Shanghai Boiler Works is presented, with special emphasis on fabrication facilities, test procedures and technological evaluations during the manufaturing process-imcluding deep driling of tubesheets, welding of tubes to tube-sheets and tube rolling tests

126
Lessons drawn from the application of probabilistic safety assessment technique in the Nuclear Research Institute. Hojny, V *Nukleon (Czechoslovakia),* No 4 3-5 (1989)

Some results are presented of reliability analyses of an emergency core cooling system for the nuclear power plant with type 213 WWER-440 reactor The results demonstrated the importance which the application of the probabilistic safety assessment techniques can have for increasing reliability of the individual plant systems The main goal, however, is safe operation of the plant as a whole This goal can be reached just by a more complex analysis of all the safety-related systems of the plant and their interrelations (author) 4 figs

127
A short description of the 440 MW reactor unit simulator of the Paks Nuclear Power Plant. Borbely, Sandor, Kiss, Istvan, Pakai, Laszlo *PAV Koezlemenyek (Hungary),* No 2, 19-27 (1989) (In Hungarian)

After a short overview of reactor simulators and operation simulation, the unit simulator of PNPP is presented The need for the simulator and the circumstances of its establishment is followed by the description of the realized PNPP 440 MW unit simulator The function and services, the simulated outages, the simulation ranges, the regimes simulated, and the fidelity of simulation are shown Finally, education and training programs by the simulator are outlined for reactor personnel (R P)

128
A new method of output power limitation in the Paks Nuclear Power Plant. Hamvas, Istvan, Kalya, Zoltan, Pos, Istvan, Elter, Jozsef, Miko, Sandor, Cserhati, Csaba *PAV Koezlemenyek (Hungary),* No 2, 29-33 (1989) (In Hungarian)

Since the installation of the Paks units, the philosophy of operational thermal output restriction established 15 years ago has been a much debated issue of reactor safety The presently used restriction practice is analyzed in the framework of changed aspects, and a completely new restriction method based on hot-spot monitoring in the reactor core is proposed This method is claimed to maintain safety of reactor units but to permit higher output in some cases (R P)

129
Guidelines of the improvement of output power limitation system valid in the Paks Nuclear Power Plant. Miko, Sandor *PAV Koezlemenyek (Hungary),* No 2, 34-37 (1989) (In Hungarian)

The limiting value restricting reactor power was defined earlier by a compu tational background that did not provide means for thermal analysis starting from the actual reactor state parameters According to this principle, preliminary analysis was made by assuming all parameters to deviate unfavourably On the other hand, experiences show that not every variable takes unfavourable value simultaneously Therefore, previous calculations yielded overpessimistic results leaving a considerable output power margin Principles for the determination of the unutilized reserves and the magnitude of the output power margin are presented (R P) 5 refs , 2 figs

130
Determination of output power margin based on thermal reliability calculations. Elter, Jozsef *PAV Koezlemenyek (Hungary),* No 2, 38-43 (1989) (In Hungarian)

Reactor power and its spatial distribution of the Paks Nuclear Power Plant have been restricted by unequality factors presently However, this method has failed to provide for the quantitative evaluation of power margin and reserves obtained with different reactor charges and operational states It is shown here that the thermal reliability of the reactor can be used as the quantity for numerical evaluation of reactor output reserves Based on this quantification, a new flexible power limitation method can be established that can be easily adopted to actual conditions (R P) 8 refs

131
Examination of radioactive corrosion products on the primary circuit surfaces of the Paks Nuclear Power Plant. Pinter, Tamas, Ormai, Peter *PAV Koezlemenyek (Hungary),* No 2, 59-62 (1989) (In Hungarian)

According to the experiences, the velocity and direction of primary circuit processes in a nuclear power reactor are influenced primarily by the primary circuit water chemistry and high-temperature pH data, and their small variations A multifactor optimization problem should be solved to find water chemistry for providing minimum out-of-core surface activity Preliminary results obtained for the relation between primary circuit corrosion product activation and water chemistry show that the activity of out-of-core surfaces is growing with the decrease of high-temperature pH value This trend complies well with theoretical models of corrosion product migration and activation in WWER-400 type reactors (R P) 7 refs 5 figs

132
Opening pressure measurement of compensator safety valves without pressure rising in pressurizer. Nemeth, Gabor *PAV Koezlemenyek (Hungary),* No 2, 63-66 (1989) (In Hungarian)

In Paks Nuclear Power Plant Hungary, main safety valves and pulsed safety valves were incorporated in the pressurizer compensator operating on the load elimination principle Operational testing of safety valves could be performed during their installation at the reactor A portable instrument was developed for in-service testing of the safety valves, in order to measure and adjust their opening pressure during operational conditions A relation was derived between the opening pressure values obtained at laboratory conditions and those under reactor operation conditions (R P) 1 fig , 1 tab

133
Operational experiences with the communication system of the Paks Nuclear Power Plant in accordance with the improvement of the management and information system. Sipos, Laszlo, Putz, Jozsef, Szuegyi, Matyas *PAV Koezlemenyek (Hungary),* No 2 70-72 (1989) (In Hungarian)

The tasks of the PNPP communication system are provision for the management and information flow related with the operation, maintenance and safety of the power production systems It consists of an operative dispatcher line network utilizing Lipka type loudspeaker intercom, and a selective paging system using Philips DP 6000 type devices for fast communication with executives and duty officers Operational experiences with both communication systems were briefly evaluated (R P) 5 refs

134
A nomogram of the economic effectiveness of the V-1 nuclear power plant life extension. Kysel, J , Skvarka, P *Zpravodaj VUPEK (Czechoslovakia),* No 3, 27-30 (1989) (In Slovak)

Economic assessment is presented of the concept variants for extending the life of the V-1 nuclear power plant after its reconstruction The starting assumption is that in 1995 the units of this nuclear power plant will be shut down for reconstruction with the aim to increase the technological safety and for updating and renewing the equipment and circuits The variant parameters in the evaluation were the shutdown time for the reconstruction, the reconstruction costs, and the year of terminating the plant operation after its reconstruction A nomogram for determining the specific annual updated costs after extending the life of the power plant units and a table listing these costs are given (P A) 2 figs , 1 tab

135
Steps to reduce consequences of major accidents of nuclear power plants (experience from IAEA course). Chytil I *Zpravodaj VUPEK (Czechoslovakia),* No 4, 51-56 (1989) (In Czech)

Low probability large accidents constitute the main hazard for nuclear power plants, as the Three Mile Island and the Chernobyl accidents showed Basic aspects of such accidents were estimated for PWR type reactors that are also used in Czechoslovakia The hypothetical course of a PWR accident is characterized and basic recommendations given which should minimize the consequences, these measures include evacuation, shelters in buildings, population resettlement, use of iodine tablets, etc (J J) 2 tabs , 3 refs

136
Development of heat supply in Czechoslovakia. Fical, S , Voboril, P *Teplo (Czechoslovakia),* 8 No 3, 14-21 (1989) (In Czech)

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

Heat extraction is envisaged for all Czechoslovak nuclear power plants in operation and under construction The turbines of the WWER-440 (2 x 220 MW) units were not designed for significant supply of steam and thus, only 60 MWth (Bohunice) or 85 MWth (Dukovany) can be obtained from them without modifications, up to 95 MWth can be obtained following minor modifications and up to 120 MWth (Mochovce) following replacement of the high-pressure body Following total reconstruction, a turbine could maximally supply 300 MWth WWER-1000 reactors with 1000 MWe turbines can supply heat of between 500 and 900 MWth, depending on the number of stages of water heating A thermal power of about 850 is envisaged for newly built units with projected applications for power and heat supply Modifications of extraction points can increase the thermal power to 1,250 MWth or to 1,500 MWth following installation of a satellite turbine Nuclear sources will gradually be linked to the existing district heating systems that today mostly use conventional sources of thermal power (J B) 4 figs , 1 tab

137
Water coolant chemistry effects on transport and radioactivation of corrosion products in nuclear power plants with WWER-440 reactors. Burclova, J , Kukucova, M , Ulicka, D , Repas, R , Dobis, L , Vagner, F , Streda, I *Jaderna Energie (Czechoslovakia),* 36 No 6, 211-216 (Jun 1990) (In Czech)

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

The effect of the primary circuit water chemistry at 300 degC on the migration and activation of corrosion products was examined for V-213 type WWER-440 reactors The dose rates in the steam generators were compared for all Czechoslovak units with reactors of this type The high-temperature pH is the most important factor in the assessment of the effect of the chemistry on the corrosion product transport at a given temperature For all units, this quantity was evaluated for the last three months before the outage The dependence of the dose rates on the high-temperature (300 degC) pH can serve as a basis for setting up recommendations for optimization of the primary circuit water chemistry The optimum pH value is approximately 7 2 This can only be achieved at boric acid concentrations lower than 2 g per kg of coolant (Z M) 3 figs , 6 tabs , 10 refs

138
Strength problems of Temelin nuclear power plant piping. Rejent, B *Jaderna Energie (Czechoslovakia),* 36 No 6, 224-231 (Jun 1990) (In Czech)

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

Strength problems are discussed for the Temelin nuclear power plant piping that serves the reactor aftercooling and interconnects the reactor, pumping station, spray tanks and cooling towers A major problem concerns the placing of the thin-walled horizontal tubes filled with water onto saddle supports Attention is also paid to the local loading capacity of the piping jacket The effect of various modifications of the saddles was examined with regard to the fact that friction in the saddles affects appreciably static strain during thermal expansion The applicability of the frictional forces to seismic vibroinsulation is discussed (Z M) 11 figs , 2 tabs , 10 refs

139
Lithium-hydroxide-accelerated oxidation of Zircaloy-4. Van Winkle, J A , Klein, A C , Maguire, M A *Transactions of the American Nuclear Society (USA),* 60 300 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The Thermal Gradient Test Facility (TGTF) constructed by Teledyne Wah Chang Albany has been designed to simulate a single, heated fuel rod under flow conditions to measure the effect of cladding oxidation on clad thermal properties Long-term pure water oxidation studies have been found to be impractical since only a limited number of data would be generated The focus of this study is to accelerate the oxidation rate of Zircaloy-4 using concentrated LiOH solutions, which will make the gathering of real-time, in situ cladding thermal property measurements in the TGTF possible Lithium hydroxide was chosen as it is already used in many pressurized water reactors to raise the pH of primary coolant

140
Application of diffusion theory methods to PWR [pressurized water reactors] analysis. Smith, K S *Transactions of the American Nuclear Society (USA),* 60 329-332 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

In-core physics analysis of pressurized light water reactors (PWRs) requires accurate predictions of three-dimensional pin-by-pin power distributions The PWR analyses must rely on diffusion theory approximation because no practical methods exist for performing routine three-dimensional pin-by-pin transport calculations Pin-by-pin diffusion calculations are also prohibitively expensive in three-dimensional geometry, and PWR analyses utilize either two-dimensional pin-by-pin models or three-dimensional advanced nodal models The purpose of this paper is to detail and contrast approximations required by pin-by-pin and nodal diffusion methods

141
Reactor plant simulation on a distributed-memory parallel processor. Park, S D , Martin, W R *Transactions of the American Nuclear Society (USA),* 60 353-355 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

A parallel algorithm for reactor plant simulation has been developed, imple mented, and tested on a distributed-memory parallel processor, the NCUBE/six hypercube parallel processor at the University of Michigan The Three Mile Island Unit 2 (TMI-2) transient was simulated with their model and a speedup of nearly 10 was obtained which is sufficiently fast to allow the simulation of the TMI-2 transient faster than real time on the NCUBE

142
Configuration management - Timely document revision. Bowen, L A , Miller, R L *Transactions of the American Nuclear Society (USA),* 60 452-453 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

An approach to document change control has been developed to ensure a timely, efficient, and cost-effective method of implementing as-built plant changes into design documents Southern California Edison Company (SCE) and Fluor Daniel are engaged in a project to routinely incorporate San Onofre nuclear generating station design changes into base documents The project encompasses two processes (1) conversion of interim design change notices (IDCNs) - those generated during the design process - and field IDCNs (FIDCNs) [or field change notices (FCNs)] - those generated during construction - into final design change notices (DCNs) and (2) incorporation of the DCNs into the base documents These processes are governed by strict procedural time limits which are specified by contract The procedures are presented in the paper

143
Fuel assembly. Nakamura, Tatsuaki, Mori, Kazuma (to Nuclear Fuel Industries Ltd , Tokyo (Japan)) Japan Patent 2-116790/A/ 1 May 1990 Filed date 27 Oct 1988 6p (In Japanese) JAPIO Also available from INPADOC

At least one outwarding protrusion is disposed on each of four peripheral sides of support lattices at the axial center of each of fuel assemblies at an identical or substantially identical height with that of a predetermined distance In addition, each of the protrusions is disposed at such a position as causing no direct interference with protrusions on the opposing side of the support lattice of adjacent fuel assemblies Accordingly, gaps at the axial center of fuel assemblies adjacent with each other in the reactor core are eliminated In this case, since the adjacent assemblies in the reactor core are brought into contact with each other in the axial center, they are supported to each other for lateral movements to provide rigidity Accordingly, vibration amplitudes of the assembly upon earthquakes are decreased to reduce the impact shocks Further, the vibrations of the assembly due to the flow of coolants are prevented Accordingly, the vibration resistance and the earthquake proofness can be improved (T M)

144
Automatic PWR type reactor control rod release mechanism. Farello, G E Naviglio, A (to ENEA Rome (Italy)) Italy Patent Application 47844A89 [1990] Filed date 13 Apr 1989 19p (In Italian) ENEA, Rome (Italy) Uff Brevetti

The automatic control rod release mechanism incorporates a heat sensor able to generate a mechanical force or torsion moment in function of reactor temperature such as to activate a mechanical actuator which allows the release of a control rod retaining device Thus the rods drop by gravity to intervene in the event of an anomalous temperature rise in the reactor vessel

145
A connection of the steam generator feedwater section of WWER type nuclear power plants. Matal, O , Sadilek, J Czechoslovakia Patent 263762/B1/ 14 Aug 1989 5p (In Czech) INIS

In the feedwater piping of each steam generator, a plate for additional water pressure reduction is inserted before the first closing valve During a steady water flow, the plate gives rise to a constant hydraulic resistance, bringing about steady reduction of the feedwater pressure, this also contributes to a stabilization of the feedwater flow rate into the steam generator The control valve thus is stressed by minimal hydrodynamic forces In this manner its load is decreased its vibrations are damped, and the frequency of failures - and thereby the frequency of the nuclear power plant unit outages -is reduced (J P) 1 fig

Graphite Moderated

146
(EGG-NPR-9151)
A review of existing gas-cooled reactor circulators with application of the lessons learned to the new production reactor circulators. White, L S (EG and G Idaho, Inc , Idaho Falls, ID (USA)) Jul 1990 Contract AC07-76ID01570 56p NTIS, PC A04/MF A01 - OSTI, GPO Dep Order Number DE91001820

This report presents the results of a study of the lessons learned during the design, testing, and operation of gas-cooled reactor coolant circulators The intent of this study is to identify failure modes and problem areas of the existing circulators so this information can be incorporated into the design of the circulators for the New Production Reactor (NPR)-Modular High-Temperature Gas Cooled Reactor (MHTGR) The information for this study was obtained primarily from open literature and includes data on high-pressure, high-temperature helium test loop circulators as well as the existing gas cooled reactors worldwide This investigation indicates that trouble free circulator performance can only be expected when the design program includes a comprehensive prototypical test program, with the results of this test program factored into the final circulator design 43 refs , 7 tabs

147
(EGG-NPR-9602)
Preconceptual design of the new production reactor circulator test facility. Thurston, G (EG and G Idaho, Inc , Idaho Falls, ID (USA)) Jun 1990 Contract AC07-76ID01570 206p NTIS, PC A10/MF A01 - OSTI, GPO Dep Order Number DE91001885

This report presents the results of a study of a new circulator test facility for the New Production Reactor Modular High-Temperature Gas-Cooled Reactor The report addresses the preconceptual design of a stand-alone test facility with all the required equipment to test the Main Circulator/shutoff valve and Shutdown Cooling Circulator/shutoff valve Each type of circulator will be tested in its own full flow, full power helium test loop Testing will cover the entire operating range of each unit The loop will include a test vessel, in which the circulator/valve will be mounted, and external piping The external flow piping will include a throttle valve, flowmeter, and heat exchanger Subsystems will include helium handling, helium purification, and cooling water A computer-based data acquisition and control system will be provided The estimated costs for the design and construction of this facility are included 2 refs , 15 figs

148
(Juel-Conf-71)
Proceedings of the workshop on structural design criteria for HTR. Breitbach, G , Schubert, F , Nickel, H (eds) (Kernforschungsanlage Juelich GmbH (Germany, F R)) Apr 1989 557p (CONF-890186-) NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan 1 feb 1989)

The papers demonstrate the status of high temperature reactor technology with regard to its realization in the nuclear power industry of various countries (FRG, USA, Japan) as well as to the development of safety rules in Germany The design criteria for HTR could be presented The criteria already determine definitely and almost completely the relevant requirements of the component rules The informations include the technical boundary conditions with regard to safety, the metallic high temperature components, a particular section dealing with the reactor pressure vessel, especially with the prestressed concrete vessel, and the structural graphite components (DG)

149
(Juel-Conf-71, pp 15-32)
Present status of MHTGR program in USA. Compiled from contributions from the MHTGR Program Team. Rittenhouse, P L Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

The U S Department of Energy (DOE) Modular High-Temperature Gas-Cooled Reactor (MHTGR) program has produced a conceptual design which has been reviewed by the U S Nuclear Regulatory Commission (NRC) The results of the review were generally favorable, and the program team has now moved into the preliminary design phase The program team consists of a nuclear island engineering (NIE) team, an energy conversion area (ECA) team, a design integration organization, and a technology development team Utility user requirements are provided by a utility organization which also participates in design and programmatic reviews/evaluations This paper will review the direction and accomplishments of each participating organization (orig)

150
(Juel-Conf-71, pp 33-47)
Present status of HTTR project in Japan. Tanaka, Toshiyuki, Saito, Shinzo (Japan Atomic Energy Research Inst , Tokai, Ibaraki (Japan) HTTR Designing Lab) Apr 1989 557p

NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

In Japan, the research and development on the HTR had been carried out for more than fifteen years as the multipurpose VHTR program for direct utilization of nuclear process heat such as nuclear steel making Recently, reflecting the change of social and energy situation and with no incentives for industries to introduce such in the near future, the JAERI has changed them for more basic 'HTTR program' to establish the HTR technology basis and upgrade them At the request of the STA the Reactor Safety Research Association has reviewed the safety evaluation guideline and started the work on the establishment of general design criteria and design code or guide for graphite and high-temperature structure of the HTTR Construction permit of the HTTR will be issued by the Government early in 1990 (orig /DG)

151

(Juel-Conf-71, pp 48-58)

HTR-situation in China. Wang, D , Xu, S (Institute of Nuclear Energy Technology, Beijing, BJ (China)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Research on HTR technology can be divided into two stages in China The first stage is from 1975 to 1978 The main aim was to look into the possibility of using HTR as a high convertor as well as a power producer Various aspects of HTR technology have been investigated The second stage started in 1984 At the beginning of the stage, the main activities include - feasibility study of using HTR in China, - R and D of HTR technology, - establishing international exchange and cooperation On the basis of these activities, HTR technology has been put into the state s high technology programme and is carring out by the institutions from the whole country International cooperation with KFA, and German industry is widened and deepened A joint programme has been set up to build a 10 MW test module HTR at INET, China (orig)

152

(Juel-Conf-71, pp 59-79)

Present status of the high temperature reactor in the Federal Republic of Germany. Nickel, H (Kernforschungsanlage Juelich GmbH (Germany, F R) Inst fuer Reaktorwerkstoffe) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

At the present time, two HTR concepts are being followed up by German companies, a small HTR (approx 200 MW_{th}) with a steel pressure vessel and a medium-sized reactor (approx 550 MW_{el}) with a prestressed concrete pressure vessel for electricity generation The special safety features of the HTR originate in the core construction of ceramic materials resistant to high temperatures and the low power density of the core An analysis of the technical safety features shows that the general activity containment concepts developed for light-water reactors (LWR) are not transferable to the HTR It was necessary to develop technical safety concepts especially for the HTR, and the result is the HTR integrity concept Licensing procedures have been established for the design, construction and commissioning of the prototype THTR 300 and invaluable experience gained A concept for licensing approval of the HTR module plants, independent of the siting of the nuclear plant, is now being investigated (orig)

153

(Juel-Conf-71, pp 113-132)

Metallurgical and physical fundamentals for the design of high temperature components. Schubert, F (Kernforschungsanlage Juelich GmbH (Germany, F R) Inst fuer Reaktorwerkstoffe) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Thermally induced, strain-controlled fatigue exposure produces, depending on temperature and strain rate, kinematic hardening or softening, whereby some proportion of the strain is creep or relaxation controlled For specific parts of the steam generator, the live steam circuit, heat exchangers and hot ducts, creep resistant steels (13 CrMoV 44) and alloys (Incoloy 800) are required to prevent time-dependent failure modes, such as creep deformation and damage, creep-fatigue, exhaustion and damage, microstructural instabilities, excessive high temperature corrosion and, for some components, loss of deformability due to neutron irradiation The metallurgical understanding behind all the failure modes are discussed The principle demands for the design of all components operating at elevated and high temperatures are the limitation of total remaining creep strains and the avoidance of localized plastic and creep strains (orig /DG)

154

(Juel-Conf-71, pp 133-158)

Load levels, stresses, failure modes and design criteria. Bieniussa, K , Reck, H (Gesellschaft fuer Reaktorsicherheit mbH (GRS), Koeln (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

The design criteria for metallic components which are to be used in the high-temperature range were described by way of example The time-dependent material behavior in this temperature range requires safeguarding against time-dependent kinds of failure, provision of time-dependent material data, and consideration of the time-dependent loads The design criteria elaborated as well as the available material data permit a design of the components that can withstand the loads for the planned operating lifetimes (orig /DG)

155

(Juel-Conf-71, pp 159-170)

Basic requirements relating to quality assurance of safety related HTR materials and components. Just, J (Rheinisch-Westfaelischer Technischer Ueberwachungs-Verein e V , Essen (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

All the quality assurance elements shown in the first figure must not be regarded separately from one another Only their planned and harmonized interaction will achieve the objective that highly stressed high temperature components will be fabricated with just as great a freedom from manufacturing defects and will be operated with as little danger of catastrophic failure as LWR components which are comparable with regard to safety and which have proven themselves over an accumulated period of more than two hundred operating years (orig)

156

(Juel-Conf-71, pp 171-184)

Non-destructive detection of flaws during manufacture and operation of components. Walte, F (Fraunhofer-Institut fuer Zerstoerungsfreie Pruefverfahren, Saarbruecken (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

The nondestructive testing (NDT) in the field of quality assurance of components in the light water reactor (LWR) technology is today very well established nevertheless not all NDT methods are transformable to the case of high temperature reactor (HTR) components The reason is the coarse grain structure of the austenitic material used in the HTR-technology in opposite to the fine grain structure in the case of ferritic material in the LWR-technology Mainly the ultrasonic (UT) testing which plays the dominant rule in the LWR inspection, is influenced by the coarse grain austenitic or nickel base alloys structure The present article analyses the influence of the coarse grain especially the dendritic structure in welds of austenitic and nickel base alloys, discusses methods and ways to detect flaws in austenitic and dissimilar welds and gives a practicable rule for NDT in the field of HTR components with a combination of ultrasonic and X-ray inspection techniques (orig)

157

(Juel-Conf-71, pp 185-205)

Material data and constitutive equations. Penkalla, H J (Kernforschungsanlage Juelich GmbH (Germany, F R) Inst fuer Reaktorwerkstoffe) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Material data and constitutive equations form the basis for the determination of design values and for the inelastic analysis of the component behaviour under complex loading conditions For metallic HTR components the materials NiCr 23 Co 12 Mo (Alloy 617), X 10 NiCrAlTi 32 20 (Alloy 800) and X 20 CrMoV 12 1 are selected The material data are obtained from the test results of different material investigations programmes Due to the high application temperatures for metallic HTR components, the main part of the material data consists of creep and fatigue properties Additional material data are the physical properties, short term properties and fracture mechanics properties The evaluated data are presented in material data sheets (orig)

158

(Juel-Conf-71, pp 243-274)

The present status of research and development works for the preparation of the high temperature design code. Muto, Y , Kaji, Y , Miyamoto, Y , Nakajima, H , Baba, O Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

A structural design code for the HTTR components is prepared in JAERI based on rules of both the Elevated-Temperature Structural Design Guide for Monju and the ASME Code Case N-47 and on material data of Hastelloy XR Research and development works have been conducted to ascertain the validity of the rules for the material and loads in the service condition of HTTR These works consist of creep tests, creep-fatigue tests, biaxial creep tests, weldment tests and component tests Test objectives, test parameters, test apparatuses and time schedules for these are introduced In addition, some test results are described A tertiary creep behavior and creep-damage criterion are discussed based on a stress controlled creep test of Hastelloy XR at 900deg C Creep-fatigue life-prediction methods are compared using fatigue-test results with both fast and slow strain rates or with both tension and compression hold times A result of bending creep fatigue test of heat transfer tubes is described (orig)

159

(Juel-Conf-71, pp 275-292)

Creep rupture characteristics in the HTGR simulated helium gas environment and their relevance to structural design. Kurata, Y , Ogawa, Y , Nakajima, H , Kondo, T (Japan Atomic Energy Research Inst , Tokai, Ibaraki Dept of Fuels and Materials Research (Japan)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Creep rupture characteristics in the HTGR simulated helium gas environment and their relevance to structural design are described for Hastelloy XR and Hastelloy XR-II, versions of Hastelloy X modified for nuclear applications The results of creep and corrosion tests in several kinds of helium environments with different impurity compositions are presented Corrosion data are analyzed to clarify the corrosion mechanism and estimate the long-term corrosion effect in the HTGR helium Creep data obtained under refined conditions make clear the effect of decarburization, carburization and oxidation on creep behaviour. On the basis of the results obtained in this study the range of gas composition is defined where decarburization and rapid carburization cannot occur for the alloys at 950deg C A data base which allows high temperature structural design for HTGR has been obtained in a standardized HTGR simulated gas (orig)

160
(Juel-Conf–71, pp 293-308)
Assessment of primary and secondary stresses for component design. Bodmann, E (Hochtemperatur-Reaktorbau GmbH, Mannheim (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

We investigated three design methods which are based on the assumption of a sufficiently ductile material It was possible to show in all three cases, that creep as a predominant or additional material effect does not influence the validity of these methods Moreover it has been shown that plasticity and creep can be treated uniformly Creep can be considered as time dependent plasticity with respect to the macroscopic effect of inelastic strain The time dependence however does not affect these design methods (orig)

161
(Juel Conf–71, pp 309-328)
Elastic and inelastic analysis of component behaviour. Seehafer, H J (Internationale Atomreaktorbau GmbH (INTERATOM), Bergisch Gladbach (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Criteria for defining the time and temperature dependent area of creep relevance are discussed by presenting corresponding creep-cross-over curves for the materials (X20 CrMoV 121, X10 NiCrAlTi 3220 NiCr22 Co12 Mo) ensuring component design against creep failure without performing any inelastic analysis The application of these creep cross over curves is restricted to normal loading conditions and does not consider possible fatigue failures where hold time effects may become significant Stress intensity factors for selected conditions and for particular material behaviour have been determined, thus enabling the performance of simplified elastic-plastic analyses Compared to ASME recommendations, significant differences depending on stress level have been identified The practical importance of this procedure is related to assessments of peak stresses due to notches and geometrical discontinuities (orig /DG)

162
(Juel-Conf–71, pp 329-345)
Significance of fracture mechanics. Schneider, K (Asea Brown Boveri AG, Mannheim (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Examples indicated that comprehensive fracture mechanics considerations may be successful on the basis of the special design criteria of HTR components Especially the proof of negligible crack growth for the relevant materials at corresponding reference temperatures should be covered by design criteria in the creep regime A stepwise procedure for the proof of rupture exclusion is proposed In the case that the conditions for rupture exclusion are not met in the first step material or component specific considerations may be applied at reasonable expense If even individual verifications do not meet the demands for rupture exclusion either service restriction and service monitoring or design review is necessary (orig /DG)

163
(Juel-Conf–71, pp 349-369)
Design criteria for prestressed concrete pressure vessels. Schimmelpfennig, K (Stangenberg, Schnellenbach und Partner Gemeinschaft Beratender Ingenieure GmbH, Bochum (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

The work concerned with the PCRVs has been focussed on topics which are not sufficiently covered by the usual codes with respect to the special structure of PCRVs and the special demands on it, and different investigations yielding a basis for such specific design criteria have been carried out Only a couple of subjects being in the fore under the aspect of defining quality enlarging design criteria for PCRVs are outlined The materials for the concrete to be used for the PCRVs are carefully selected (DG)

164
(Juel-Conf–71, pp 370-384)
Design criteria for liners of concrete vessels. Oberpichler, R (Stangenberg, Schnellenbach und Partner Gemeinschaft Beratender Ingenieure GmbH, Bochum (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

The composite liner for HTR-500 can be expected highly safe against failure as long as the global integrity of the composite structure 'steel liner + anchorage + concrete structure' can be guaranteed Therefore the reliability of the composition between the liner plate and the concrete structure is of great importance for designing the liner anchor system That means the anchorage concept with regard to type, size and arrangement of the anchors, the connection of the anchors to the liner plate and the embedding of the anchors themselves in the concrete are responsible for the composition between liner and concrete Thus, design criteria for analysing and fabricating a composite liner have been worked out which predominantly have to guarantee a safe anchorage Standard proposals for several design details concerning material, analysis, construction and testing of the composite liner have been worked out (orig /DG)

165
(Juel-Conf–71, pp 385-403)
Special features of the design of pressure vessel closures and heat insulations. Pschowski, J (Hochtemperatur-Reaktorbau GmbH, Mannheim (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Based on realistic design data different anchor concepts were analysed for the transitional areas between the penetration liner and the vessel closure Making use of the basis safety criteria an integrity concept has been devel oped in connection with the double closure design which meets the specific HTR-500 conditions The effect of the failure on the load carrying behav iour of the PCRV and on the leak tightness of the liner was evaluated Strength analyses were carried out and the fracture mechanics of the failure mechanism was evaluated (DG)

166
(Juel Conf–71, pp 404-441)
The HTR-module pressure vessel unit, design criteria and safety philosophy. Neumann, G , Heidt, K , Dumm, K , Rothfuss, H (Siemens AG Unternehmensbereich KWU, Erlangen (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

The pressure vessel unit of the HTR module can make use of experience gained from the construction and oper ation of reactor coolant system components of light water reactors Therefore proven techniques and accepted codes and standards are available for all aspects such as design material manufacture inspection, operational monitoring and inservice inspection At the same time the safety concept of light water reactor primary system components against catastrophic failure which is characterized by multiple redundancies, can be applied in full This means that the in tegrity of the pressure vessel unit of the HTR module is assured for the total lifetime (orig /DG)

167
(Juel-Conf–71, pp 467-479)
Materials behaviour and design values. Haag, G (Forschungszentrum Juelich GmbH (Germany, F R) Inst fuer Reaktorwerkstoffe) Apr 1989 557p NTIS (US Sales Only) PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR, Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

In structural design criteria for high temperature reactors graphite has to be regarded as a material of its own which is significantly different from other construction materials To guarantee the quality of graphitic materials, it is necessary to verify the properties of raw materials to survey the production process and to control the final product properties Designing graphitic reactor components one has to take into account problems that may arise from corrosion and from fast neutron radiation effects Consequently, physical properties such as dimensional stability, Young's modulus, tensile strength, thermal conductivity, thermal expansivity and coefficient of irradiation induced creep have to be observed as a function of neutron fluence in irradiation experiments (orig)

168
(Juel-Conf–71, pp 480-492)
Design methods and criteria for graphite components. Schmidt, A (Hochtemperatur-Reaktorbau GmbH, Mannheim (Germany, F R)) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186–)

From Workshop on structural design criteria for HTR Julich (Germany, F R) (31 Jan - 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Criteria have been established on the basis of the THTR licensing proce dure and a material research and development programme for the design of graphite internals of an HTR-500 The components were classified into classes according to their special functions, the loads applied to the components were classified into load levels depending on their occurrence probability For evaluating the stresses the Weibull theory was used Allowable failure probabilities are defined rather than allowable stresses This evalua tion method can be applied to any stress state (orig)

169
(Juel-Spez–565)
Irradiation behaviour of advanced fuel elements for the helium-cooled high temperature reactor (HTR). Nickel, H (Forschungszentrum Juelich GmbH (Germany, F R) Inst fuer Reaktorwerkstoffe) May 1990 22p (CONF-900634–) NTIS (US Sales Only), PC A03/MF A01 Order Number DE91717161

From 7 world ceramics congress ceramics today–tomorrow's ceramics, Montecatini Terme (Italy) (24-30 Jun 1990)

The design of modern HTRs is based on high quality fuel A research and development programme has demonstrated the satisfactory performance in fuel manufacturing irradiation testing and accident condition testing of irradiated fuel elements This report describes the fuel particles with their low-enriched UO_2 kernels and TRISO coating, i e a sequence of pyrocarbon, silicon carbide, and pyrocarbon coating layers, as well as the spherical fuel element Testing was performed in a generic programme satisfying the requirements of both the HTR-MODUL and the HTR 500 With a coating failure fraction less than $2x10^{-5}$ at the 95% confidence level, the results of the irradiation experiments surpassed the design targets Maximum accident temperatures in small, modular HTRs remain below 1600deg C, even in the case of unrestricted core heatup after depressurization Here, it was demonstrated that modern TRISO fuels retain all safety-relevant fission products and that the fuel does not suffer irreversible changes Isothermal heating tests have been extended to 1800deg C to show performance margins Ramp tests to 2500deg C demonstrate the limits of present fuel materials A long-term programme is planned to improve the statistical significance of presently available results and to narrow remain ing uncertainty limits (orig)

170
Berechnung der Nachzerfallsleistung der Kernbrennstoffe von Hochtemperaturreaktoren mit kugelfoermigen Brennelementen. Dokumentation und Erlaeuterungen (Decay heat power in nuclear fuels of high-temperature reactors with spherical fuel elements; documentation and illustration). Berlin (Germany, F R), Beuth (May 1990) 22p (In German) (DIN-25485(suppl 1))

This additional sheet contains information for DIN 25 485, but no additional standardized regulations (orig)

171
Berechnung der Nachzerfallsleistung der Kernbrennstoffe von Hochtemperaturreaktoren mit kugelfoermigen Brennelementen (Decay heat power in nuclear fuels of high-temperature reactors with

spherical fuel elements). Berlin (Germany, F R); Beuth (May 1990) 13p (In German) (DIN–25485)

The standard supplies the basis for the calculation of the decay heat power of non-recycled nuclear fuel The following parts are taken into account the contribution of the fission products from nuclear fission, the contribution of the actinides, the contribution of isotopes which are produced by neutron capture in fission products It shows the local production of decay heat power relative to the thermal fuel power during operation The process of calculation standardized here has the advantage of calculating the decay heat power with an accuracy comparable to summation programs, without needing expensive computer programs and extensive data libraries (orig /HP)

172

Development of large heat supply systems with long distance heat transported in the chemically combined state. Fedyaev, A V pp 1-37 of Scientific and technical progress in district heating and cogeneration Levin, L I , Benenson, E I , Dlugosel'skii, V I , Pilyagin, V F , Khristich, L M , Gribov, V B , Smirnov, I A , Fedyaev, A V Chur (Switzerland), Harwood Academic Publishers (Feb 1990) 141p

Experience in developing and constructing radically new systems for the generation, transmission and utilization of nuclear power in the national economy is generalized Consideration is given to methods used for research, and the technical and economic prerequisites for the development of integrated power supply systems based on high temperature nuclear reactors and the technology for transmitting thermal power in a chemically combined state (orig)

173

Critical experiments at very high temperature reactor critical assembly (VHTRC). Akino, F , Yamane, T , Yasuda, H , Kaneko, Y (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp IV 13-IV 22 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409–)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

In order to verify the calculation accuracy related to the neutronic design of HTTR (High Temperature Engineering Test Reactor), critical experiments have been conducted using the critical assembly VHTRC (Very High Temperature Reactor Critical Assembly), which has pin-in-block fuel The VHTRC-1 core was made by loading rods which contained fuel compacts of the coated particles of the 4% enriched uranium The following measurements were carried out on the VHTRC-1 core 1) critical mass, 2) reactivity worths of HTTR mockup control rod and burnable poison rod, 3) neutron flux distribution 4) temperature coefficient of reactivity and 5) kinetic parameter Calculations were performed with the SRAC code system using the nuclear data based on the ENDF/B-IV The agreements between calculations and experiments were fairly good for most experimental items except the value of kinetic parameter The results obtained satisfy the accuracy requirments for fundamental nuclear design of HTTR

174

The Soviet RBMK: where do we go from here?. Adamov, E O *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 431, 33-36 (Jun 1990)

After implementing a series of safety measures, the Soviet nuclear engineers now believe that there is no possibility of a repeat of the Chernobyl accident sequence and there is some discussion of further development of the RBMK concept The concept, with these safety measures, is examined and the advantages of the channel type uranium-graphite design noted (author)

175

Specific power bound of a finite-time closed Brayton cycle. Chih Wu *International Journal of Ambient Energy (UK),* 11 No 2, 77 82 (Apr 1990)

The purpose of this paper is to define a finite-time closed Brayton cycle and apply it to optimize an indirect gas-cooled nuclear reactor plant Practical engineering heat engine power optimization usually takes the form of determining minimum heat exchanger area per unit net power output or minimum cost per unit power output, rather than determining the maximum cycle efficiency The objective function of the optimization described in this paper is specific power, power output per total heat exchanger surface area The specific power output of a real indirect gas-cooled nuclear reactor plant coupled with its heat source and sink is analyzed It is found that there is an upper bound on the specific power output of the nuclear power plant. This bound can guide the evaluation of existing real plants or influence design of future power plants (author)

176

Results of 21st regular inspection of Tokai Power Station, Japan Atomic Power Co. (Agency of Natural Resources and Energy, Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 24-25 (Mar 1990) (In Japanese)

The 21st regular inspection of Tokai Power Station was carried out from December 20, 1988 to August 30, 1989 The parallel operation was resumed on August 1, 1989, 225 days after the parallel off The facilities which were the object of inspection were the reactor proper, reactor cooling system, measurement and control system, fuel facilities, radiation control facilities, waste facilities and emergency power generation system On these facilities which were the object of inspection, the appearance, disassembling, leak, function, performance and other inspections were carried out As the result, it was found that a part of the thermocouple cable trays for measuring reactor exit gas temperature fell, but other abnormality was not observed The works related to this regular inspection were accomplished within the range of the limit of dose equivalent based on the relevant laws The main reconstruction works carried out during the period of this regular inspection were the replacement of 32 steam side safety valves, the replacement of 6th bellows in No 1 and No 2 main gas ducts, and the repair of thermocouple cable trays in the reactor (K I)

177

Vibrational characteristics of superheater elements in the steam separator-superheater for the RBMK-1500 reactor. Makarov, I I , Grebennikov, V N *Ehnergomashinostroenie (USSR),* No 6, 37-40 (Jun 1989) (In Russian)

Results of tests conducted to determine the values of eigenfrequencies and logarithmic decrements of vibrations of basic elements of a steam superheater for a NPP with the RBMK-1500 type reactor are presented

Vibrational characteristics of heat exchange tubes and spacing grids are determined

178
Review of 21 years of power operation at the AVR experimental nuclear power station in Juelich. Ziermann, E *Nuclear Engineering and Design (Netherlands),* 121 No 2, 135-142 (Jul 1990)

Our experimental nuclear power station is the first plant with a helium-cooled high-temperature reactor which uses ball-shaped fuel elements Despite the imperfections and difficulties resulting from this fact, it was possible to construct and commission the plant within a relatively short space of time The plant was run successfully for power operation for a little more than 21 years It has been shown that the reactor possesses excellent safety characteristics and that the radiation exposure for the personnel and for the environment was very low Operation of the plant presents no problems and makes no great demands on the operating personnel All the faults that occurred could essentially be eliminated with our own plant crew The condition of the principal components does not show any signs of changes due to ageing at any point On the basis of our operational experience and the previous running of the plant, no restrictions for further use of the plant can be derived today either (orig)

179
Results of experiments at the AVR reactor. Gottaut, H , Krueger, K *Nuclear Engineering and Design (Netherlands),* 121 No 2, 143-153 (Jul 1990)

The most important experiments at the AVR reactor and their results are discussed This choice illustrates the significance of the 'AVR experiment' for high-temperature reactor development in Germany and for the construction and operation of future HTR projects (orig)

180
Construction and operating experience with the 300-MW THTR nuclear power plant. Baeumer, R , Kalinowski, I , Roehler, E , Schoening, J , Wachholz, W *Nuclear Engineering and Design (Netherlands),* 121 No 2, 155 166 (Jul 1990)

After a long construction period which was mainly induced by changing safety and licence requirements, the THTR 300 nuclear power plant was successfully commissioned after starting electricity generation in November 1985 Evalution of the operating experience reveals absolutely positive results the principal design data have been achieved, the principal design of the large pebble bed reactor was confirmed Shut down of the plant after the generation of 2 9 million MWh and 423 full power days is due to financial risk of the prototype plant (orig)

181
Status and aims of the research and development program for high-temperature reactors. Kirch, N , Grosser, E D , Herzberger, K H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 167-171 (Jul 1990)

Modern HTR designs in Germany like the HTR-Module and the HTR 500, rely on 30 years of research and development, on more than 20 years of successful AVR operation and testing, and on the experience gained from THTR Due to this broad developing program, HTR technology in Germany has reached a high level of maturity Nevertheless, as in any other technical area the development is going on steadily Novel materials and techniques, higher general safety requirements, and the advanced reactor designs inspire and require additional investigations With regard to the components and systems, the work is concentrated on simpler and more economical solutions In the safety field it is necessary to achieve a quality for the proofs sufficiently high for licensing procedures and a clear demonstration of the key safety features (orig)

182
Experience gained from the EVA II and KVK operation. Harth, R , Jansing, W , Teubner, H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 173-182 (Jul 1990)

This paper deals with tasks, structure and experience gained from the two helium test facilities, EVA II and KVK, which are part of the main tools within the framework of component development for a high-temperature reactor Special techniques (internal insulation, cooling by using cold gas) for the components became necessary to separate high temperatures (950deg C) from the pressure-loaded component parts The question of helium atmosphere and its interaction with high-temperature materials could be sufficiently investigated Well-developed oxide layers were obtained on the helium side Excellent reliability and flexibility of the facilities have been obtained The following components have been tested in due time and at moderate costs Steam reformers, helium heat exchangers, steam generators, hot gas valves, hot gas ducts, blowers In addition to the experimental results practical experience gained with helium technology is of essential importance for further work relating to HTR development and design (orig)

183
Design and qualification of components for high-temperature reactors. Dumm, K , Theymann, W , Decken, C B von der *Nuclear Engineering and Design (Netherlands),* 121 No 2, 193-198 (Jul 1990)

The status of development of the main components belonging to the HTR-M and the HTR-500 is described Depending on the different design features of both reactor types, the main design activities and R and D efforts were directed to different components But the activities together cover the whole range of components necessary for both projects For instance, the graphitic core structures and main circulators of the HTR-500 and the HTR-M are of similar design, and the hot gas ducts and steam generators developed for the HTR-M are applicable to the HTR-500 In addition, the combined activities of KFA, HRB and Interatom were found to be very successful and economic (orig)

184
Development of advanced HTR fuel elements. Nabielek, H , Kuehnlein, W , Schenk, W , Heit, W , Christ, A , Ragoss, H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 199-210 (Jul 1990)

Spherical fuel elements with coated particles are the essential constituents of the integrity of a pebble-bed reactor, particularly with respect to fission product retention While all gas-cooled reactors exhibit an extremely low primary circuit activity, recent developments have reduced the source term to virtually zero This entails an enormous effort in the characterization of the fuel behaviour in manufacture, during irradiation, and in accident testing With modern SiC-coated particles it can be shown that all radiologically relevant fission products are completely retained inside the silicon carbide layer of intact fuel particles The dominant source term for fission product release will therefore, be the small number of particles with defective coatings Permanent damage to the fission product retaining SiC layer can only start in accidents involving temperatures significantly beyond 1600deg C (orig)

185

Development and qualification of materials for structural components for the high-temperature gas-cooled reactor. Nickel, H , Schubert, F , Breitling, H , Bodmann E *Nuclear Engineering and Design (Netherlands),* 121 No 2, 183-192 (Jul 1990)

This paper summarizes the materials evaluation work which has been done for the advanced HTR projects Essentials of the materials information needed for the design of prestressed concrete pressure vessels (PCPV), steel pressure vessels, graphite incore structures, and metallic heat components inside the pressure vessel are discussed The main emphasis is given to the metallic high-temperature components which are exposed in the temperature regime in which all properties are sincerely temperature and time dependent For those components and for graphitic side reflectors the methods of analysis and proof of integrity of the components during the total operation time are provided (orig)

186

Designer himself throws light upon high-temperature reactor. Andriesse, C D *Energiespectrum (Netherlands),* 14 No 4, 114-115 (Apr 1990)

THe high-temperature reactor is one of the alternatives for the now predominantly employed water-reactors In a recently published book designer Rudolf Schulten outlines his concept In this article the book is reviewed (author) 1 ref 1 fig

187

Thermodynamic investigations of passive decay heat removal from HTR cores and component behaviour. Rehm W , Altes, J , Barthels, H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 211-218 (Jul 1990)

This paper describes core heatup transients for the THTR-300 and HTR-Module and the knowledge obtained with the experimental facility LUNA-HTR for the study of natural convection characteristics of different HTR primary circuits In consideration of thermodynamic similarity principles, transients with heat removal via natural convection from a pebble-bed into the primary circuit were simulated, as well as calculated with the code THERMIX-CRAY-2D On the basis of experiments the code is furthermore used for the analysis of natural decay heat dissipation in the depressurization case of the primary circuit Studies on HTR safety have indicated the great significance of processes resulting during temperature stressing of the concrete pressure vessel for the sequence and consequences of accidents, particularly those with unrestricted core heat-up In the course of the accident the vessel is slowly heated up to very high temperatures Over the last ten years the behaviour of PCRVs of different HTR (THTR, HTR-500) plants during core heat-up accidents has been analyzed Besides the PCRV itself, also the top reflector suspension is experimentally investigated (orig)

188

Fission product behaviour and graphite corrosion under accident conditions in the HTR. Katscher, W , Moormann, R , Verfondern, K , Decken, C B von der, Iniotakis, N , Hilpert, K , Christ, A , Lohnert, G , Wawrzik, U *Nuclear Engineering and Design (Netherlands),* 121 No 2, 219-225 (Jul 1990)

This paper primarily gives an overview of methods and data in source term estimations for the HTR with pebble bed core For medium size HTRs the risk dominating accidents are tied to core heat-up events, where a significant portion of the fission product inventory may be released from the coated fuel particles Here the research mainly is focused on temperature-induced coated particle failure and the interaction of metallic fission products with the core graphite For small HTRs, with their limitation of maximum temperatures below coated particle failure limits, core heat-up accidents virtually play no role with respect to source terms Here the risk is dominated by accidents like water ingress or rapid depressurization which may lead to a partial release of fission products accumulated on primary circuit surfaces like the steam reformer Deposition of fission products and remobilization under the conditions mentioned above are predominant research areas It can be expected that the ongoing and planned improvements of models and data base, in particular for the medium size HTR, will result in a further reduction of the already low source terms A principal possibility for core degradation and hence destruction of fission product barriers is graphite corrosion caused by massive air ingress The research effort in this field as well as for graphite corrosion during water ingress accidents is described in Part B of this paper From the viewpoint of risk for this type of accident no significant contribution to that of present reactor concepts was found (orig)

189

Status of the high-temperature reactor (HTR) - applications. Candeli R , Arndt, E , Barnert, H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 249-258 (Jul 1990)

The high temperature reactor with pebble bed core is a universal heat source with a high degree on passive safety features, which supplies electric power and process steam as well as process heat for industrial applications at economical interesting conditions Since the high temperature reactor is especially suitable for applications within the process heat markets, the HTR will contribute widely to the reduction of coal dioxide emission also in this market area (orig)

190

Technical design features and essential safety-related properties of the HTR-module. Lohnert, G H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 259-275 (Jul 1990)

An HTR-Module power plant consists of standardized reactor units with a power rating of 200 MW each, so that the special, inherent safety features of small high-temperature reactors are also applicable to power plants of any desired power rating The main design features of the reactor and vital components are given Special emphasis is layed on a comprehensive survey of design basis accidents and the corresponding accident doses in the environment It is shown that these doses are much lower than the maximum allowable values given in the German Radiation Protection Ordinance even if it is postulated that the installed filters of the reactor building are not available Finally, a summary of essential properties and design principles of the HTR-Module is given These properties and principles were fully ap proved and accepted by the advisory experts (TUeV) of the German Licensing Authorities in their final assessment report on the conceptual design of the HTR-Module (November 1989) (orig)

191
Design status of the HTR-500 power plant. Arndt, E , Schoening, J , Wachholz, W *Nuclear Engineering and Design (Netherlands),* 121 No 2, 277-286 (Jul 1990)

The HTR 500 has been developed as a nuclear power plant for medium-sized and large electrical grids, either as a base load plant or for load-following applications according to the requirements of the grid Furthermore the HTR 500 is well suited for cogeneration or supply of process steam for industrial applications (orig)

192
The 20-MW gas-cooled heating reactor: Upgrading the GHR 10. Schmitt, H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 287-291 (Jul 1990)

The gas-cooled heating reactor (GHR) matches the requirements of heating reactor systems in terms of safety, simplicity and economy The GHR is designed on the basis of the proven technology of the gas-cooled high-temperature reactor The main benefits are derived directly from the technical and safety related HTR characteristics Due to these characteristics, the GHR plants can be sited near urban centers without any risk to the population (orig)

193
Basic risk analyses for high-temperature reactors. Kroeger, W , Mertens, J , Wolters, J *Nuclear Engineering and Design (Netherlands),* 121 No 2, 299-309 (Jul 1990)

For medium-sized HTRs (e g HTR-500) of current design failure of active systems for decay heat removal, resulting in core heatup, clearly dominates the risk and leads to the largest releases of radioactive nuclides into the environment For small-sized HTRs (e g HTR-Module) temperature-induced releases from the fuel are insignificantly low for all types of accident, plate out activities on the steam generator surfaces remobilized in the course of water ingress accidents can be regarded as the main contribution to the comparatively small source term The largest releases are so low for all HTR concepts that early health effects can be ruled out in any case, including no evacuation For small HTR plants even late cancer effects need practically not to be expected A comparison with licensed released values has shown that the applicable current requirements are met by all HTR concepts examined However, small HTRs especially offer an additional potential for compliance with more stringent safety requirements, 'taking the fear out of hypothetical accidents', by limiting maximum releases Incidentally, the classically defined 'risk' to the population from both plants is generally very low (orig /HP)

194
High temperature reactor core physics and reactor dynamics. Wolf, L , Scherer, W , Giesser, W , Feltes W *Nuclear Engineering and Design (Netherlands),* 121 No 2 227 240 (Jul 1990)

Core physics and reactor dynamics of high temperature reactors (HTR) are well understood in their basic principles due to the high standards of models and computer code systems available In what follows an overview is given concerning the basic properties of HTR core design and dynamics code systems The validation of these systems is accomplished using critical experiments and power plant operation experience The final verification of the HTR-inherent safety properties calls for further development of the dynamic models including hypothetical accidental situations such as the ingress of very large amounts of water or air into the primary circuit combined with the total failure of all control systems (orig)

195
Storage and final disposal of spent HTR fuel in the Federal Republic of Germany. Kirch, N , Brinkmann, H U , Bruecher, P H *Nuclear Engineering and Design (Netherlands),* 121 No 2, 241-248 (Jul 1990)

The spent fuel treatment concept for HTR in the FRG is based on direct disposal of the fuel in a salt dome repository Due to high burnup and good in situ fuel utilization, direct disposal offers economic advantages, especially for low enriched uranium fuel Besides, the safety requirements can be met by simple techniques due to the special features of the HTR fuel element coated particle fuel, stabilized in a graphite matrix with absence of any metal and the low heat production per volume unit give favourable preconditions for intermediate storage and safe disposal Techniques for the intermediate storage are available and practised with AVR and THTR fuel For final disposal, emplacement in boreholes, 300-600 m in depth, using simple packaging, was chosen as reference, similar to the reference concept for heat generating, medium-active waste So far, the results of both the development and the experimental test programme underline the chosen concept (orig)

196
Concept licensing procedure for an HTR-module nuclear power plant. Brinkmann, G , Will, M *Nuclear Engineering and Design (Netherlands),* 121 No 2, 293 298 (Jul 1990)

In April 1987 the companies Siemens and Interatom applied in the West German state of Lower Saxony for a concept licensing procedure to be initiated for an HTR-Module nuclear power plant In addition to a safety analysis report, numerous additional papers were submitted to the authorized experts In April 1989 proceedings were suspended for political and legal reasons By this time both the fire protection report and the plant security concept report had been completed The safety concept review was continued by order of the Federal Minister for Research and Technology The draft safety concept report was completed in July 1989 The final version was completed at the end of 1989 (orig)

197
11th international conference on the HTGR and 10th IAEA technical committee meeting on gas-cooled reactor technology, safety and siting. Suransky, F , Podest, M , Pinkas, V *Jaderna Energie (Czechoslovakia),* 36 No 6, 236-239 (Jun 1990) (In Czech)

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

Conclusions are summarized from the 11th international conference on the HTGR and from the 10th IAEA technical committee meeting on gas-cooled reactor technology, safety and siting, which were held in the USSR in June 1989 It is concluded that high-temperature reactors are promising sources of high potential heat and, owing to their specific physical and technical characteristics, exhibit a high degree of inherent safety and are favorable from the ecological point of view A number of approaches exist to the exploitation of the high-potential heat Problems, however, remain to be solved with respect to the commercial application of high-temperature reactors which have to compete with existing technologies (Z M)

198
Introducing the high-temperature reactor into the market - status and strategy. Baust, E , Weisbrodt, I A *Nuclear Engineering and Design (Netherlands),* 121 No 2, 311-315 (Jul 1990)

The pebble bed high temperature reactor has been taken to the threshold of commercialization in more than 30 years of development On the basis of the experience gained with the 15-MW AVR reactor and the THTR 300, marketable plant sizes (HTR 500, HTR Module, gas cooled heating reactor - GHR) have been developed for the electricity and heat market, which are now available for future energy supply The high-temperature reactor represents a reasonable supplement to the proven light-water reactors and is particularly suited for export to developing countries and industrial threshold countries in view of its technical and safety characteristics and its wide range of applications in the electricity and heat market ABB and Siemens have decided to pursue the future HTR product development and marketing activities in a long-term strategy by the joint HTR GmbH company There is a worldwide interest in the HTR which is manifested by several international cooperation agreements (orig)

199
Status of design of the HTR test module China. Steinwarz W Xu Yuanhui *Nuclear Engineering and Design (Netherlands),* 121 No 2, 317 324 (Jul 1990)

Within the frame of German-Chinese R and D cooperation, initiated at the end of 1988, the HTR-Module technology will be implemented in the People's Republic of China as an element of future energy supply Furthermore, the partners agreed a common further development of this technology As a first step, the design of a 10-MW(th) experimental reactor was begun, which is characterized by the main features of the commercial-sized HTR-Module With this so-called Test Module to be errected close to Beijing, it is planned, as one main topic, to demonstrate the fuel element integrity in the temperature range of 1600deg C (orig)

200
Examination and exchange of fuel element centering pins at Biblis A nuclear power station. Wolf, M , Poetz, F *VGB Kraftwerkstechnik (Germany, F R),* 70 No 9, 778-781 (Sep 1990) (In German)

During the 1989 unit inspection fuel element centering pins of the material Inconel X-750 were changed for the first time at Biblis A Nuclear Power Station In preparation for this, within 6 months, a suitable exchange device was developed and proved Following a review of the damage to the fuel element centering pins in 1300 MW plants, the reason for the development of an exchange device is briefly described Finally, by means of design features, the testing- and exchange concept and the development of the device are illustrated (orig)

201
Reaction of nuclear-grade graphite with low concentrations of steam in the helium coolant of an MHTGR. Richards, M B *Energy (Oxford) (UK),* 15 No 9, 729-739 (Sep 1990)

The fuel elements for the MHTGR are manufactured from nuclear-grade graphite and have core-residence times of about 1000 days that are determined from considerations based on nuclear design criteria To ensure structural integrity and minimize the risk to plant investment, a potential limitation of fuel element life may result from graphite corrosion in the high-temperature regions of the core ($T \simeq$ 1100degC) This corrosion is caused primarily by reactions of graphite with the low concentrations of steam (0 01 - 0 1 ppm) that are normally present in the helium coolant Economical opera tion of the reactor requires that the steam concentrations be maintained at such low levels that corrosion rates will not impact the normal fuel cycle In this paper, we develop a graphite-corrosion model and address this important practical problem (author)

202
Cooling structure for the lower surface of reactor core support plate. Tsuji, Nobumasa (to Fuji Electric Co Ltd , Kawasaki, Kanagawa (Japan)) Japan Patent 2-116791/A/ 1 May 1990 Filed date 27 Oct 1988. 4p. (In Japanese) JAPIO Also available from INPADOC

In a HTGR type reactor in which the lower surface of a reactor support plate is cooled by supplying coolants between the reactor core support plate and a cover plate, gap between the reactor support plate and the cover plate is partitioned horizontally In this case, a rectification plate having a gap is disposed at the periphery of a main coolant pipe which passes through the reactor core support plate and no gap at the periphery of an auxiliary coolant pipe which passes through the reactor core support plate is disposed By such a constitution, since the coolants do not flow to the lower surface of the reactor core support plate from periphery of the auxiliary coolant pipe but all of them are flow to the lower surface of the reactor support plate from the periphery of the main coolant pipe radially and uniformly, the entire reactor core support plate can be cooled uniformly (T M)

Otherwise Moderated or Unmoderated

203
(EUR-12578, pp 623-632)
Fuzzy diagnosis. Watanabe, K (Power reactor and nuclear fuel development corp Oarai Ibaraki, (JP) Oarai engineering center) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

Studies have been made on fuzzy diagnosis using inverse problem solutions of the fuzzy relational equation of ao R=b, where a is the failure vector R the fuzzy relation matrix and b the sympton vector Four phases of analyses were carried out in this study First, fault tree analysis was undertaken to investigate what kind of causes produce fall of water level in a steam drum of ATR (Advanced Thermal Reactor), which is heavy-water-moderated boiling-water-cooled pressure-tube-type reactor Next, simulation for 100 seconds was executed to determine how plant parameters respond to an occurrence of a transient induced by the cause Third, the simulation data was analysed utilizing an autoregressive model From this analysis, a total of 36 coherency functions up to 0 5 Hz in each transient were computed among nine important and detectable plant parameters, that is neutron flux, flow rate of coolant, steam and feed water, water level in the steam drum, pressure and opening area of control valve in a steam pipe, feed water temperature and electrical power Last, the inverse problem of the fuzzy relational equation was solved Relation matrices were adjusted from 0 00 to 1 00, after nine membership functions following the Gussian distribution for the symptom vector were estimated from correlation values of the coherency functions

204
(IAEA-TECDOC-556, pp 83-94)
CANDU used fuel handling and storage. Nakagawa, R K (Atomic Energy of Canada Ltd Mississauga, ON (Canada) CANDU Operations) Jun 1990 151p NTIS (US Sales Only), PC A08/MF A01 OSTI, INIS Order Number DE91607408 (CONF-8904225-)

From IAEA Technical Committee meeting on decontamination of transport casks and spent fuel storage facilities Vienna (Austria) (4 7 Apr 1989)

In Decontamination of transport casks and of spent fuel storage facilities

A review is made of CANDU fuel handling and storage systems using the CANDU Model 6 reactor as typical, including a description of the on-power refuelling system Basically two fuelling machines are used to refuel the selected fuel channel (horizontal pressure tube) of the CANDU pressurized heavy water cooled reactor Used fuel storage bays are normally constructed for a ten year operating capacity As the bays become full, auxiliary fuel storage bays or another storage means becomes necessary Dry storage of irradiated fuel has been developed as an alternative and has been successfully employed on the decommissioned Gentilly-1 and Douglas Point Nuclear Generating Stations (author) 8 figs

205
The use of subcritical multiplication for the improvement of conversion ratio in CANDU lattices. Dastur, A R Mao A C , Chan, P S W (Societe Francaise d Energie Nucleaire (SFEN) 75 - Paris (France)) pp I 64-I 72 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

A high-burnup, high-utilization, once-through fuel cycle, that is based on the natural uranium CANDU fuel cycle, is presented The concept exploits the on-power fuel management capability of CANDU to shift fuel from under-moderated, high-conversion lattices to well-moderated lattices where the neutron spectrum is better suited to burn plutonium Fuel enrichment is avoided by using the well-moderated lattice to drive the under-moderated lattice, which remains sub-critical but makes a substantial contribution to the reactor power Natural uranium utilization is 3 2 times better than in a conventional CANDU and 4 4 times better than in an LWR Spent fuel volume is reduced to 31% of present CANDU production and is 146% of LWR The main drawbacks of this concept are related to the low power density of the fuel in the subcritical regions

206
Simulation of reactivity devices in PHWRS - validation against low power start-up experiments in NAPS-1. Srivenkatesan, R , Jagannathan, V , Krishnani, P D , Degwekar, S B , Jain, R P , Modak, R S , Srinivasan, K R (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp IV 57-IV 64 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The Unit 1 of Narora Atomic Power Station (NAPS-1) Narora, India, achieved criticality on March 12, 1989 A series of physics experiments with various reactivity devices were done in subcritical, low power critical and dynamic power decay conditions of the reactor These were analysed with our computer code systems The paper describes briefly the methods of analysis, experiments and comparison of the calculated results with measurements

207
Studies in the Zed-2 critical facility of reactivity coefficients for Candu cores fueled with plutonium fuels. Jones, R T , Griffiths, J (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 73-VI 82 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Measurements and calculations relevant to the understanding of coolant-void and coolant/fuel-temperature reactivity coefficients of Pu-containing fuel in CANDU reactor lattices have been made The measurements used few-rod substitution techniques in cores fuelled mainly with natural uranium Calculations were made with the lattice-code WIMS-AECL (using a data library based on ENDF/B-5) and a reactor core code (CONIFERS) Conclusions drawn are that the calculations overestimate the reactivity effect of voiding the coolant and underestimate a positive reactivity contribution to the fuel/coolant temperature coefficient

208
Core management of the MOX fuel loaded heavy water reactor. Yoshitake Shiratori, Takehide Deshimaru, Mitsuo Matsumoto, Kuniyoshi Saito (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VI 23-VI 33 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Fugen has been operating safely and smoothly for ten years using mainly plutonium-uranium mixed oxide (MOX) fuels and much operation data has been accumulated during that time Fugen's core management has been established, and its core characteristics have been evaluated through the operation Core analysis code POLESTAR has been developed by PNC and it has enough functions for core management This code has contributed reliable and economical operation of Fugen, since its accuracy has been checked and some modifications of the code have been done by comparing its calculation results with various measured data

209
Clearing debris from the core at Argentina's Atucha 1 PHWR. *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432 31 33 (Jul 1990)

Damage arising from a broken guide tube and fuel/coolant channel at Argentina s Atucha 1 Pressurized Heavy Water Reactor kept the plant off line for 16 months Remnants of the tube and other debris had to be cleared from the reactor vessel through 120mm holes 11m above the working area The repair programme and the design of the manipulators involved are described (author)

210
Results of investigating cause of defective contact of timer relay for emergency low oil pressure in ATR 'Fugen' Power Station, Power Reactor and Nuclear Fuel Development Corporation. (Science and Technology Agency, Tokyo (Japan) Nuclear Safety Bureau) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135, 22 (Mar 1990) (In Japanese)

In the ATR Fugen Power Station, during the adjustment operation at the rated output as the final stage of the 8th regular inspection on November 6, 1989 due to the signal 'Main steam stop valve close the reactor automatically stopped As to the cause of the signal Main steam stop valve close being made the plant process data before and after the automatic stop of the reactor were examined as the result it was presumed that it occurred due to the temporary defective contact of the timer relay for emergency low oil pressure in the turbine control system Therefore, the relay concerned and three timer relays of the same type were examined on the contact resistance at the contact points, the analysis of the contact point surfaces, and the analysis of the structural materials It was presumed that the concentration of the silicone gas generated from silicone rubber in the relay became high, silicon oxide was formed by the decomposition of silicone gas due to contact point arc, and adhered to the contact point surface, thus the contact resistance increased The new timer relays in which silicone rubber is not used were adopted (K I)

211
Fuel assembly for pressure tube type reactor. Wakabayashi, Toshio (to Power Reactor and Nuclear Fuel Development Corp , Tokyo (Japan)) Japan Patent 2 116788/A/ 1 May 1990 Filed date 26 Oct 1988 4p (In Japanese) JAPIO Also available from INPADOC

For providing a effect of burnable poisons without worsening local power peaking, it is effective to dispose conventional fuel rods incorporated with burnable poisons uniformly to the outermost layer of a fuel assembly in which fuel rods are disposed concentrically in a multilayered state However, since the irradiation is applied to the outermost layer with high neutron fluxes in this case, the burnable poisons are eliminated rapidly and excess reactivity suppression effect does not last sufficiently Then, according to the present invention, pellets of a dual layer structure comprising a pellet containing burnable poisons disposed at the center and nuclear fuel materials composed of plutonium-uranium mixed oxides or uranium oxides coated therearound are disposed In view of the above, it is possible to obtain a fuel assembly which does not increase the suppression of excess reactivity at the initial stage and does not promote the elimination of burnable poisons, without lowering the local power peaking (T M)

Breeding

212
(AERE-TP-1354)
Developments in modelling the effect of aerosol on the thermal performance of the Fast Reactor cover gas space. Ford I J , Clement C F (UKAEA Harwell Lab (UK) Theoretical Physics Div) Mar 1990 23p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607717

The sodium aerosol which forms in the cover gas space of a Fast Reactor couples the processes of heat and mass transfer to and from the bounding surfaces and affects the thermal performance of the cavity This report describes extensions to previously separate models of heat transfer and aerosol formation and removal in the cover gas space, and the linking of the two calculations in a consistent manner The extensions made to the theories include thermophoretic aerosol removal, radiative-driven redistribution in aerosol sizes, and the side-wall influence on the bulk cavity temperature The link between aerosol properties and boundary layer saturations is also examined, especially in the far-from-saturated limit The models can be used in the interpretation of cover gas space experiments and some example calculations are given (author)

213
(EGG-M-89448)
"Tightly coupled" simulation utilizing the EBR-II LMR: A real-time supercomputing and AI environment. Makowitz, H , Barber, D G , Cordes, G A , Powers, A K , Scott, R Jr , Ward, L W , Sackett, J I , King, R W , Lehto, W K , Lindsay, R W (EG and G Idaho, Inc , Idaho Falls ID (USA)) [1990] Contract AC07-76ID01570 6p (CONF-900343-7) NTIS, PC A02/MF A01 - OSTI, GPO Dep Order Number DE91001942

From Supercomputing in nuclear applications, Mito City (Japan) (12-16 Mar 1990)

An integrated Supercomputing and AI environment utilizing a CRAY X-MP/216, a fiber-optic communications link, a distributed network of workstations and the Experimental Breeder Reactor II (EBR-II) Liquid Metal Reactor (LMR) and its associated instrumentation and control system is being developed at the Idaho National Engineering Laboratory (INEL) This paper summarizes various activities that make up this supercomputing and AI environment 5 refs , 4 figs

214
(EUR-12578, pp 271-292)
Discussion on an informative system set-up for the registration and processing of reliability data on FBR components in view of its application to design and safety studies and plant exploitation improvement. Righini, R , Sola, P G , Zappellini, G (ENEA Rome (IT)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156-, CSNI-159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

This report describes the set-up and management activities carried-out by ENEA-VEL in collaboration with NIER in the development of a reliability data bank on fast reactor components, this data bank consists of an informative system implemented on the IBM 3090 computer of the ENEA centre of Bologna starting from the software of the CEDB set-up by CCR Euratom of Ispra for the registration of reliability data on thermal reactor components This report will contain a detailed description of all the modules (engineering operating etc) provided in the informative system and of the modifications introduced by ENEA in order to adapt them to the peculiarities of the fast reactors and to increase its flexibility a short description of the available data processing methods will be also included It will be followed by a comparison between the results obtained applying the classical methods and the particular ones set-up by ENEA this comparison will be useful to demonstrate the importance of the method applied in order to obtain significative reliability processed data This report will be also useful to show the importance of the set-up data bank in the improvement of the component design and of the plant safety and exploitation with particular reference to the research of the critical areas and to the definition of the best inspection and maintenance programs

215
(EUR-12669)
LMFBR safety criteria and guidelines for consideration in the design of future plants. (Commission of the European Communities Luxembourg (Luxembourg)) 1990 50p NTIS (US Sales Only), PC A03/MF A01

For many years the Commission of the European Communities has been conducting activities aimed at the progressive harmonization of safety requirements and criteria applied to nuclear installations in the Community These activities cover thermal and fast reactors This publication represents a major achievement in reaching this goal The document, which has been prepared in the framework of activities of the CEC fast-reactor safety working group (SWG) consists of safety criteria and guidelines for fast reactors It represents the common view of all EC Member States which have a fast-reactor programme or are interested in fast-reactor development The criteria and guidelines are structured according to different types of possible faults, such as core reactivity faults, general cooling faults, subassembly faults, faults outside the core and causes external to the station Only those events are considered which are in the design basis of current fast-reactor projects Proposed measures or guidelines to satisfy the criteria are based on the present knowledge and proven technology

216
(JAERI M-90 116)
Fabrication of uranium-plutonium mixed nitride fuel pins (88F-5A) for first irradiation test at JMTR. Suzuki Yasufumi Iwai Takashi Arai Yasuo Sasayama Tatsuo Shiozawa Ken ichi Ohmichi Toshihiko Handa Muneo (Japan Atomic Energy Research Inst Tokyo (Japan)) Jul 1990 51p NTIS (US Sales Only) PC A04/MF A01 Order Number DE91723351

A couple of uranium plutonium mixed nitride fuel pins was fabricated for the first irradiation tests at JMTR for the purpose of understanding the irradiation behavior and establishing the feasibility of nitride fuels as advanced FBR fuels The one of the pins was fitted with thermocouples in order to observe the central fuel temperature In this report, the fabrication procedure of the pins such as pin design, fuel pellet fabrication and characterizations, welding of fuel pins, and inspection of pins are described, together with the outline of the new TIG welder installed recently (author)

217
Mixed oxide fuels with minor actinides for the fast reactor. Pilate, S , Wouters, R de, Evrard, G , Wiese, H W Wehmann U (Societe Francaise d Energie Nucleaire (SFEN) 75 - Paris (France)) pp I 73 I 82 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The addition of a few percents of Neptunium-237 to the fuel of a large fast breeder like EFR is found advantageous to the core operation, as it allows to reduce the plutonium enrichment, to lower the burnup reactivity loss (and simplify the design of control rod absorbers) and to increase the plutonium breeding gain The two latter effects are also found in the case of plutonium ageing (long storage) leading to the accumulation of Am-241 If required, the use of uranium from reprocessing does not bring noticeable penalties on reactor operation The efficiency of the FBR as an actinide burner is compared to that of U- and MOX LWRs and found equal or better, depending on the isotope

218
Recent objectives and trends in fast reactor core design. Thornton D E J (Societe Francaise d Energie Nucleaire (SFEN) 75 - Paris (France)) pp II 1 II 10 of The physics of reactors operation design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF 900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

In the past, many advantages have been claimed for fast reactors over thermal ones, a principal one being their economy in uranium use, often quoted as 60 times better than thermal reactors, though 140 times may be more reasonable This advantage has recently seemed less important, particularly in countries holding large uranium reserves, and safety-related advantages have been pursued, whilst elsewhere benefits in generation cost have been sought at the expense of breeding performance Whilst this is justifiable in the early development of the system when development of reliable engineering is the prime goal, the pursuit of high fuel efficiency will have to be resumed for the fast reactor to achieve its fundamental goal of extending the world nuclear energy resource to the maximum extent possible

219
An evaluation of LMR design options for reduction of sodium void worth. Hill, R N , Khalil, H (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp II 19-II 33 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Systematic analyses of alternative methods for reducing the sodium void worth of plutonium fueled liquid metal reactors (LMRs) have been performed The focus of this study is on core de signs of recent interest in the U S LMR program, i e designs in the 450 to 1200 MWt size range that make use of metal alloy fuel Design alternatives are investigated which encompass changes in composition, geometry, and blanket arrangement A self-consistent evaluation is made of the void worth reduction achievable by various methods and the associated core physics performance trade-offs The performance penalties (e g reduced breeding efficiency and increased burnup reactivity loss and fissile mass requirement) caused by design changes which significantly reduce the void worth are quantified, and the relative merits of each design option are assessed

220
Core optimization studies for the European fast reactor EFR. Wehmann U K Wouters R de Sun derland R Sztark H (Societe Francaise d Energie Nucleaire (SFEN) 75 - Paris (France)) pp II 34-II 43 of The physics of reactors operation de sign and computation Volume 1 Paris (FR) Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF 900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23 26 Apr 1990)

For the European Fast Reactor EFR a first consistent core design has been worked out during the conceptual design phase which lasted up to March 1990 This paper describes in its first part the main features of the EFR core design and the boundary conditions and criteria which had to be fulfilled The second part is devoted to some special design aspects Efforts to re duce the sodium void effect and the flexibility to adapt the breeding gain to a required target value are outlined It is shown that axial heterogeneous cores are offering interesting core design improvements and that optimized fuel management schemes have the potential to increase the fuel discharge burnup without exceeding a given maximum burnup

221
R and D programme in France in support of design studies devoted to new FBR's core concepts. Bergeonneau, P , Cabrillat, J C , Gibiat, F , Martini, M , Picard, E , Rippert, D (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp II 44-II 53 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

A specific R and D programme is underway in the CEA in order to improve future FBR's core performances and safety level The objective of this programme is mainly to get a better knowledge and consequently to reduce the associated uncertainties of the neu tronic performances of the core and of the fuel elements behaviour In each field a large part is devoted to the axial heterogeneous fuel concept This programme is principally based on an experimental approach the CONRAD programme in the MASURCA facility for the core physical performances and concerning axial heterogeneous fuel, the irradiation of ax het subassem blies in the FBR PHENIX core

222
The neutronic and fuel cycle performance of interchangeable 3500 MWth metal and oxide fueled LMRs. Fujita, E K , Wade, D C (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp II 54-II 74 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

This study summarizes the neutronic and fuel cycle analysis performed at Argonne National Laboratory for an oxide and a metal fueled 3500 MWth LMR These reactor designs formed the basis for a joint U S /European study of LMR ATWS events The oxide and metal core designs were developed to meet reactor performance specifications that are constrained by requirements for core loading interchangeability and for a small burnup reactivity swing Differences in the computed performance parameters of the oxide and metal cores, arising from basic differences in their neutronic characteristics, are identified and discussed It is shown that metal and oxide cores designed to the same ground rules exhibit many similar performance characteristics, however, they differ substantially in reactivity coefficients, control strategies, and fuel cycle options

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Study on high performance axially heterogeneous core for a 1000MW(e) LMFBR. Mari Yano, Kunihiro Itoh, Takeshi Hojuyama, Nariaki Uto, Toshikazu Takeda (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp II 75-II 87 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

High performance axially heterogeneous core (AHC) concepts for a large LMFBR are studied by varying the internal blanket material and configuration from the view-points of safety characteristics and fuel integrity A new AHC concept proposed in this paper may be called as Island AHC It has been revealed that this Island AHC has the following superior core performances compared with a conventional AHC (1)Reducing the sodium void re activity, (2) Reducing the power peaking factor, (3) Reducing the fast neutron fluence, and (4)Improving the fuel behavior related to fuel integrity

224
New results of the use of recent nuclear data files for U-233 and TH-232 on doubling time characteristics for a typical FBR. Ganesan, S , Ramanadhan, M M , Gopalakrishnan, V , Menon, P V K , Lee, S M (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp III 1-III 8 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The group constants of processed nuclear data of U-233 and Th-232 from recent evaluated nuclear data files are intercompared U-233 data were processed from ENDL-84/V, JENDL-2 and ENDF/B-IV and Th-232 data were processed from ENDF/B-V (Rev 2) and JENDL-2 Comparison of these later data with the standard data in use in Kalpakkam for FBR design are presented The pre-processing and processing of data were carried out using the IAEA codes LINEAR, RECENT and SIGMA1 and the Kalpakkam codes REX1 REX2 and REX3 The U-233 and Th-232 data thus obtained were used to study the characteristics of a 500 MWe FBR in terms of breeding ratios fissile inventories and doubling times

225
A review of progress with the Janus programme of fast reactor shielding benchmark experiments. Calamand, D Curl I J (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 1-V 11 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The JANUS experimental programme on the NESTOR reactor at Winfrith was defined to provide validation data for neutron penetration in LMFBR shield components The programme commenced in 1987 and the current programme should be completed in 1991 This paper summarizes the scope of the JANUS programme of shielding benchmark experiments and reports the progress to date in both the experimental work and the analyses

226
Measurements and analysis of gamma-ray energy deposition in a critical assembly containing a central simulated diluent. Wouters, R de Calamand, D Granget G , Cleri, F D Hondt, P Stanculescu A (Societe Francaise d Energie Nucleaire (SFEN) 75 - Paris (France)) pp V 12-V 21 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The radial distribution of gamma dose rate deposited in Fe has been measured in the assembly BALZAC-DE1, built in the zero-power MASURCA fast breeder facility The assembly contained a central singularity simulating a steel/sodium diluent such as used in SUPERPHENIX, surrounded by a PuO_2-UO_2 driver zone Experimental techniques involved calibrated ionization chambers and thermoluminescent dosemeters Theoretical analyses carried out by three laboratories with their own data and methods, led to the conclusion that the current libraries underestimate the photon sources in plutonium fuelled zones and that the use of recently recommended photon yields for fission in Pu would worsen the discrepancies between calculation and measurement The ratio C/E in the central steel zone ranges from 0 75 to 0 97

227
Experimental investigations of physical characteristics of FBR heterogeneous cores at BFS facility. Matveyenko, I P , Matveyev, V I , Belov, S P , Bobrov, S B , Tcherny, V A , Efimenko, V F , Kochetkov, A L (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 22-V 31 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation Paris (France) (23-26 Apr 1990)

Experimental investigations of FBR heterogeneous core physical characteristics have been carried out at the critical assemblies of the BFS facility Both radial and axial heterogeneous cores have been investigated Critical dimension, fission reaction rate distributions, sodium void reactivity effect control rod worths have been measured during the experiments The experimental results are compared with the results of calculations obtained by the methods and codes usually employed in fast reactor design

228
Experimental study of physics characteristics related to neutron flux distribution in an axially heterogeneous LMFBR. Iijima, S , Obu, M , Ohno, A , Sakurai, T (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 32-V 40 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

By the insertion of the fertile material zone at the middle of the core, neutron flux distribution in an axially heterogeneous LMFBR differs to that in a conventional homogeneous LMFBR The distribution, which shows different shape with change of neutron energy, mutually relates to the typical physics characteristics such as a power-flattening, a reduction of positive sodium void worth and a constant differential worth of control rod These relationship was investigated using the experiment and the calculation results in FCA assembly XII-1 and XIII-1

229
The CONRAD programme: experiments and analysis for an axially heterogeneous core in the MASURCA facility. Cabrillat, J C , Gauthier, J C , Martini, M , Palmiotti, G , Salvatores, M , Soule, R , West, J P , Sacre, G , Helm, F , Polch, A (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 41-V 50 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-27 Apr 1990)

The CONRAD programme at the MASURCA facility of the CEA Cadarache (FRANCE) is performed as part of the European collaboration on fast breeder reactor in support of the EFR project (European Fast Reactor) A brief description of the complete programme is given in the paper along with its justification The first criticality was obtained in June 1989 An analysis of the first results has been made and is also included

230
Experiments and analysis for large conventional fast reactors in ZPPR-18/19. Brumbach, S B , Collins, P J , Carpenter, S G , Grasseschi, G L (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 63-V 73 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Experiments in ZPPR 18 and 19 provided physics data for large (1100 MWe) conventional LMRs Measurements focused on the spatial distributions of fission rates and control worths Experimental results were obtained for the fundamental to first-harmonic eigenvalue separation In the outer core of ZPPR-18, fuel was arranged in sectors with uranium about the y axis and plutonium about the x axis In ZPPR-19 the fuel was rear ranged to give a uniform outer core distribution Fission rate C/E values were essentially invariant with radius through the plutonium zones The control worth C/E values were influenced by their proximity to the uranium sectors

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SFINX: Soviet-French integral experiment on measuring the capture and fission at Masurca and BFS. Doulin, V A , Mikhailov, J M , Mozhaev, V K, Soule, R , Bertrand, P , Brocart, A , Granget, G , Marquette, J P (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 74 V 83 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The SFINX experiment was aimed at the comparison of experimental procedures used at the MASURCA and the BFS critical assemblies for measuring the ratio of the ^{238}U and ^{239}Pu average fission cross-sections to the ^{235}U average cross-section (F8/F5 F9/F5) and of the ^{238}U average capture cross-section to the ^{239}Pu average fission cross section (C8/F9) As part of the calibration of the measurements F8/F5 and F9/F5 were also measured in a thermal column To obtain C8/F9, absolute measurements of capture rates in ^{238}U and fission rates in ^{239}Pu and ^{235}U were carried out The measure ments were made in September 1987 at the MASURCA facility (FRANCE) in the BALZAC 1 critical assembly and in the thermal column of the HARMONIE facility In April 1989 these measurements were complemented by joint measurements of the ^{239}Pu absolute fission rate at the BFS 55-1 critical assembly (USSR)

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Calculational and experimental studies of physical characteristics of the BN-600 reactor with the modified core. Matveyenko, I P , Matveyev, V I , Zvonarev, A V , Efimenko, V F , Evdokimov, V P , Kolyzhenkov, V A , Roslyakov, V F , Suvorov, V D , Tchernyl, B A (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VII 46-VII 55 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Measurements of basic characteristics and a calculational analysis of the measurements results were carried out after the change-over of the BN-600 reactor to the modified core In the paper the analysis results are presented for the power distribution and for the control rod worths In the experiments the power distribution was determined by measurements of gamma radiation intensity after fresh subassemblies were irradiated in the core for a short time at the low power level Control rod worths were measured using an inverse kinetic technique The calculational analysis was made on the basis of 3D diffusion theory code using BNAB cross sections set

233

Analysis and interpretation of the neutronic experimental results at the super-phenix re-start up in 1989. Bergeonneau, P , Cabrillat, J C , Vanier, M , Zaetta, A , Languille, A , Sztark, H (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VII 1-VII 9 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

This paper describes the neutronic experiments which were realized at the SUPER-PHENIX re-start up during the first semester of 1989 and the present status of their interpretation They consisted in two series of tests first, zero power experiments aiming at measuring the core reactivity value, evaluating by an appropriate simulation the reactivity brought by the handling error taken into account in the core design, measuring the control rods worth and the reactivity effects caused by fuel-diluent subassemblies substitutions, second, core reactivity balances and coefficients measurements all along the increase of the reactor thermal power up to the nominal value

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Common lessons drawn from different laboratories analyses of super-phenix start-up experiments. Cabrillat, J C , Salvatores, M , Carta, M , D'Angelo, A , Giese, H , De Wouters, R , Newton, T , Harrison, P , Sztark, H *(Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France))* pp VII 10-VII 20 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Measurements issued from the SUPER-PHENIX start-up experiments have been analysed by the different partners within the European Community with their own data and methods Common lessons can be drawn from the different analyses and recommendations made on the definition of the characteristics of a common European formulaire and in the actions in support of its qualification

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Some lessons learned from the evaluation of superphenix-1 start-up experiments. Wehmann, U K , Cabrillat, J C (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VII 21-VII 28 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

During the start-up phase of Superphenix-1 several neutronphysics and thermalhydraulics measurements have been performed, with which the safe and reliable operational behaviour of the reactor core has been demonstrated Additionally, these experiments are essential for the validation of core design data and methods This paper describes the results of some evaluations which have been performed with

Interatom methods They are related to the critical mass absorber worth values burnup reactivity loss and radial distribution of fuel subassembly power The evaluations have shown the ade quacy of the methods which also are being applied for the design of the European Fast Reactor EFR

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Comparison of calculation and measurements of reaction rates in the outer regions of superphenix. Robinson, P J , King, D C , Wall, S J , Cabrillat, J C , Calamand, D , Palmiotti, G (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp VII 29 VII 38 of The physics of reactors operation, design and computation Volume 1 Paris (FR) Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF 900409–)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

This paper brings together the results of two analyses of the Superphenix start-up measurements in the radial shield Two different methods of calculation have been used based on SN transport theory (the PROPANE DO formulaire with the BISTRO code) and adjusted diffusion coefficients (the code SNAPSH using ADC method D) The PROPANE DO formulaire was found to give good agreement with the measured thermal flux both in the prediction of radial attenuation and in the reproduction of the axial profile The ADC method was found to underestimate the attenuation There were difficulties in reproducing the flux and neutron spectrum at the breeder/shield interface with both methods This detailed comparison with measured values will assist in the formulation of recommended calculational routes for analysis of fast reactor shielding problems, an example of which has been the proposal of revised uncertainties for the PROPANE DO formulaire

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Core physics parameters measured in the carbide fueled fast breeder test reactor. Lee, S M , Reddy, C P , Sathiamoorthy, V , Shankar Singh, R (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp VII 39-VII 45 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409–)

From International conference on the physics of reactors operation, design and computation Paris (France) (23-26 Apr 1990)

A summary is presented of the main observations noted during the low power physics commissioning experiments of the mixed carbide fueled Fast Breeder Test Reactor The initial core loading and approach to criticality are described followed by an overview of the measurements pertaining to control rod worths, subassembly worths and the feedback reactivity coefficients related to coolant temperature, flow and cover gas pressure The methods of analysis have been mainly based on two dimensional cylindrical geometry diffusion theory with hexagonal geometry and transport theory corrections by subsidiary calculations In general the agreement between predicted values and measurements is considered satisfactory

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Analysis of RVACS tests for commix validation. Tzanos, C P , Pedersen, D R *Nuclear Engineering and Design (Netherlands),* 121 No 1, 59-67 (Jul 1990)

Two RVACS tests, A and B, were analyzed to support the validation of COMMIX and the design of Advanced Liquid Metal Reactors (ALMRs) The RVACS test facility provides a scaled simulation of the decay heat removal paths of a pool ALMR during the operation of the passive decay heat removal systems with the reactor simulated with electrically heated rods Test A was characterized by a power input of 50 kW, natural convection in the sodium pool, forced RVACS air circula tion, sodium pool overflow, and a heat up period of 8 h In test B the power was 64 kW and the sodium pool heatup led to overflow at $\simeq$ 3 h from the beginning of the transient In test A, after 7 5 h, the system reached near steady state with a temperature difference between the bottom and top of the pool of 96deg C In both tests the vertical temperature gradient in the hot pool remained small throughout the transient while the cold pool exhibited a significant stratification The sodium temperature variations in the radial and azimuthal directions were insignificant The COMMIX predictions for the sodium pool temperatures and the air outlet temperatures were in good agreement with measurements (orig)

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PHOENICS offers cost-effective computation of transient fluid dynamics. *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 52-53 (Jul 1990)

Assessing what happens in a reactor during transient operation is fraught with problems PHOENICS mathematical models developed by CHAM Consultancy/BV Neratoom have been used successfully to predict thermal-hydraulic behaviour in the SNR 300 Fast Breeder Reactor under both steady state transient conditions (author)

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Super phenix fast reactor fuel storage canister substitution: Ansaldo (Italy) design. Parodi, B *Energia Nucleare (Milan) (Italy),* 7 No 1, 45-47 (Jan-Apr 1990) (In Italian)

Civil design characteristics are com pared between the previous fuel storage canister which sprung a sodium leak (the exact cause of which is still under investigation) and its replacement component Whereas the previous canister was designed to store new fuel elements and cool spent fuel in sodium kept at 200 degrees C, the new version is being designed only for fuel loading and unloading operations and will not contain sodium Its installation is to take place during reactor operation In addition to providing the key dimensions of the replacement canister, this paper outlines the principal design steps employed, computer codes and validation procedures

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Development of HT-9 for liquid-metal reactor components. Johnson, G D *Transactions of the American Nuclear Society (USA),* 60 286-287 (1989) (CONF-891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Alloy HT-9 is being used for both duct and cladding applications in advanced liquid-metal reactor (LMR) experiments This tempered martensitic steel was selected for use as an LMR core component material primarily because of its excellent resistance to radiation-induced swelling Experiments conducted in the Fast Flux Test Facility (FFTF) at 410°C and exposures in the range of 150 to 175 displacements per atom (dpa) have shown that Ht 9 exhibits only a 0 2 to 0 3% increase in volume Cold-worked austenitic steels exhibit volumetric increases of 20 to 30% at 410°C Alloy HT-9 is being used for a series of fuel pin experiments in the FFTF, and these tests have achieved a burnup of 175 MWd/kg metal and a fluence of 25 × 10^{22} n/cm^2 (E > 0 1 MeV) without fuel pin breach The high confidence placed in HT 9 is based on a wide series of in- and ex-reactor experiments Test results for these experiments are summarized in this paper

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A liquid-metal reactor core demonstration experiment using HT-9. Ethridge, J L , Walter, A E *Transactions of the American Nuclear Society (USA),* 60 289-290 (1989) (CONF-891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The Core Demonstration Experiment (CDE) is an ongoing multiassembly fuel and blanket irradiation experiment in the US Department of Energy s Fast Flux Test Facility (FFTF) located near Richland, Washington The structural material (cladding and duct) for the irradiation assemblies is HT-9 The objective of the program is to demonstrate the long-lifetime (3-yr) capability of this particular fuel system with its attendant economic benefits The purpose of this paper is to present programmatic results obtained to date related to the performance of HT-9 cladding and duct components

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Irradiation creep behavior of HT-9. Puigh, R J *Transactions of the American Nuclear Society (USA),* 60 294-295 (1989) (CONF-891103)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

For the past 20 yr, significant effort has been placed in the investigation of neutron irradiation effects on the mechanical properties of candidate structural materials for advanced liquid-metal reactors (LMRs) More recently, this effort has been extended to the neutron irradiation environment envisioned for future fusion reactor concepts Specifically, materials testing and development for these programs has focused on application temperatures in the 300 to 700°C range and extension of the neutron exposures as far as practical Dimensional stability in these candidate materials is a highly desirable characteristic, which has been measured for many different classes of candidate structural materials In these early studies, the martensitic alloy HT-9, has been shown to exhibit superior resistance to dimensional changes due to void swelling Due to these early results, HT-9 was chosen as a leading materials candidate for high fluence structural material applications by the LMR program An important materials property for both the LMR and fusion programs is in-reactor creep behavior This report describes the most recent insights into the irradiation creep behavior of HT-9 Data have been accumulated on several different composition and thermomechanical treatments of HT-9 to neutron fluences > 120 displacements per atom

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Microstructural evolution of martensitic steels during fast neutron irradiation. Maziasz, P J *Transactions of the American Nuclear Society (USA),* 60 297-299 (1989) (CONF-891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Irradiation of martensitic/ferritic steels with fast neutrons (E > 0 1 MeV) to displacement damage levels of 30 to 50 displacements per atom (dpa) at temperatures of 300 to 500°C produces significant changes in the as-tempered microstructure Dislocation loops and networks can be produced, irradiation-induced precipitates can form, the lath/subgrain boundary structure and the thermal precipitates produced during tempering can become unstable, and if helium is present, bubbles and voids can form These microstructural changes caused by irradiation can have important effects on the properties of this class of steels for both fast breeder reactor and magnetic fusion reactor applications The purpose of this paper is to compare reactor-irradiated and long-term thermally aged 9Cr-1MoVNb specimens to distinguish effects due to displacement damage from those caused by elevated-temperature exposure alone

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Feedback components of a Pu-fueled compared to a U-fueled 900 MWt LMR. Meneghetti, D , Kucera, D A *Annals of Nuclear Energy (UK),* 17 No 7, 353-361 (1990)

The feedback components of the regional contributions of the power reactivity decrement (PRD) and of the temperature coefficient of reactivity for a 900 MWt homogeneous UPu10Zr-fueled Na-cooled fast reactor are calculated These PRD components are also separated into power dependent and power-to-flow dependent parts The values of the PRD components and of the temperature coefficient components are compared with corresponding quantities of an analogous U10Zr-fueled reactor The effects of these values upon quantities useful in the interpretations of inherent safety characteristics of metal-fueled Na-cooled fast reactors are presented (author)

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Irradiation performance of metallic fuels. Pahl, R G , Lahm, C E , Porter, D L , Batte, G L , Hofman, G L *Transactions of the American Nuclear Society (USA),* 60 304-305 (1989) (CONF-891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Argonne National Laboratory has been working for the past 5 yr to develop and demonstrate the Integral Fast Reactor concept The concept involves a closed system for fast reactor power generation and on-site fuel reprocessing, both designed specifically around the use of metallic fuel The Experimental Breeder Reactor II has used metallic fuel for all of its 25-yr life In 1985, tests were begun to examine the irradiation performance of advanced-design metallic fuel systems based on U-Zr or U-Pu-Zr fuels These test have demonstrated the viable performance of these fuel systems to high burnup The initial testing program is described in this paper The performance data base to date has shown that metallic fuel systems can perform reliably to high burnups over a wide range of design conditions The data base is continually being strengthened to support performance modeling efforts and development of advanced reactor systems such as General Electric Company's PRISM

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Development of the metallic fuel performance code SESAME. Kobayashi, Tsuguyuki, Kinoshita, Motoyasu, Hattori, Sadao, Iwai, Takashi, Tsuboi, Yasushi, Ishida, Masayoshi, Ogawa, Shinta, Saito, Hiroaki *Transactions of the American Nuclear Society (USA),* 60 305-306 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26 30 Nov 1989)

The potential of metallic fuel for the liquid metal fast breeder reactor (LMFBR), enlightened by the pioneering efforts of Argonne National Laboratory is expected to provide them with a new breakthrough in the areas of passive safety and an economically competitive fuel cycle In Japan the Central Research Institute of the Electric Power Industry (CRIEPI) first recognized its possibilities and has been conducting a wide range of feasibility studies covering both core design and pyroreprocessing, including laboratory experiments The SESAME (Simulating Evaluation System of Al loyed Metallic Element) code has been developed to predict steady-state irradiation performance of metallic fuels based on simple experimental correlations for EBR II fuels Results are described in the paper

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In situ observation of axial irradiation growth in liquid-metal reactor metal fuel. Cramer, E R , Pitner, A L *Transactions of the American Nuclear Society (USA),* 60 306 307 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Effects of the rapid early-in-life expansion of metal fuel were measured in an irradiation experiment in the Fast Flux Test Facility (FFTF) This important performance/design information was obtainable through the unique combination of a dimensionally stable FFTF oxide core and the calibrated proximity instrumentation associated with the test These results delineate the time dependence of metal-fuel swelling and provide quantitative estimates of the magnitude of axial fuel swelling in full length metal-fuel assemblies Final posttest examination results will define actual fuel column growth levels

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Fuel performance in JOYO. Kashihara, H , Shikakura, S , Yokouchi, Y , Shibahara, I , Matsushima, H , Nomura, S *Transactions of the American Nuclear Society (USA),* 60 309-310 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The Japanese experimental fast reactor JOYO MK-II has been operated at 100 MW since March 1983 as a fuel and material irradiation facility Many types of fuel irradiation tests for the prototype reactor MONJU and demonstration type fast breeder reactor FBR have been conducted by using four types of irradiation rings, including an instrumented test assembly (INTA) The paper discusses experience with driver fuel irradiation and MONJU-type fuel irradiation tests

250
Status of advanced LMR [liquid metal reactor] fuel development in Switzerland. Alder H P , Stratton, R W , Ledergerber, G Botta, F *Transactions of the American Nuclear Society (USA),* 60 310 311 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The Swiss Federal Institute for Reactor Research, now called the Paul Scherer Institute (PSI), has continued to develop advanced fuels for the fast reactor [liquid-metal reactor (LMR)] as well as for light water reactors (LWRs) using gelation techniques Fabrication development, irradiation testing, postirradiation examination, and fuel behavior modeling form an important part of the activities The results to date show that gelation techniques are a viable method for producing advanced ceramic fuels and that at least in the sphere-pac form, good behavior can be expected in the reactor

251
Improvements in the fabrication of metallic fuels. Tracy, D B , Henslee, S P , Dodds, N E , Longua, K J *Transactions of the American Nuclear Society (USA),* 60 314-315 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Argonne National Laboratory (ANL) is currently developing a new liquid-metal-cooled breeder reactor known as the Integral Fast Reactor (IFR) The IFR represents the state of the art in metal-fueled reactor technology Improvements in the fabrication of metal fuel, discussed in this paper, will support ANL-West's (ANL-W) fully remote fuel cycle facility, which is an integral part of the IFR concept

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Simulation of the injection casting of metallic fuels. Nakagawa, Tomokazu Ogata, Takanari, Tokiwai, Moriyasu *Transactions of the American Nuclear Society (USA),* 60 315-316 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

For the fabrication of metallic fuel pins, injection casting is a preferable process because the simplicity of the process is suitable for remote operation In this process, the molten metal in the crucible is injected into evacuated molds (suspended above the crucible) by pressurizing the casting furnace Argonne National Laboratory has already adopted this process in the Integral Fast Reactor program To obtain fuel pins with good quality, the casting parameters, such as the molten metal temperature, the magnitude of the pressure applied, the pressurizing rate, the cooling time, etc , must be optimized Otherwise, bad-quality castings (short castings, rough surfaces, shrinkage cavities, mold fracture) may result Therefore, it is very important in designing the casting equipment and optimizing the operation conditions to be able to predict the fluid and thermal behavior of the castings This paper describes methods to simulate the heat and mass transfer in the molds and molten metallic fuel during injection casting The results obtained by simulation are compared with experimental ones Also, appropriate casting conditions for the uranium-plutonium-zirconium alloy are discussed based on the simulated results

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Fabrication and in-pile performance of helium-bonded mixed nitride fuel pins at the beginning of

life. Richter, K , Blank, H *Transactions of the American Nuclear Society (USA),* 60 319 (1989) (CONF-891103–)
From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)
The studies described in this paper are part of a program whose aim is to optimize the performance of advanced fast breeder reactor fuel The preparation fabrication and irradiated behavior of mixed uranium-plutonium nitrides with low oxygen and carbon contents are described The material was prepared by the carbothermic reduction of compacted mixtures of uranium and plutonium oxides with carbon black under an atmosphere of nitrogen and mixed nitrogen-hydrogen at 1,500 to 1 700°C for 5 to 20 h Pellets of uranium plutonium nitride with 84 ± 2% theoretical densities were prepared by three routes (1) conventional method (C) using milling pressing and sintering techniques, (2) pressing and sintering of granulated material (G), and (3) direct pressing (P) of the clinker of nitride obtained from the carbothermic reduction of the oxides The sintered pellets were made up into pins that were then irradiated in a TRIO capsule in the epithermal neutron flux (Cd screen) of the 45-MW thermal materials testing high flux reactor at Petten the Netherlands Some initial nondestructive postirradiation examination was carried out at the reactor facility No marked differences were observed in the behavior of fuel obtained by the three fabrication process Studies of ceramic fuels with densities in the 80 to 85% range have not hitherto been carried out Information has been obtained about the mechanisms of the changes that occur when the material is subjected to in pile irradiation

254
Helium- and sodium-bonded mixed carbide fuels - In-pile behavior and head-end gaseous oxidation. Coquerelle, M , Ronchi, C , Blank H *Transactions of the American Nuclear Society (USA),* 60 320 (1989) (CONF-891103–)
From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)
This paper reports on a detailed analysis that has been carried out on hyperstoichiometric uranium-plutonium carbide fuel ($U_{0.8}Pu_{0.2}C$), which has operated under helium- and sodium-bonded conditions up to 12.5 at% burnup in a fast flux reactor The first aspect of the analysis of the fuel was the swelling due to solid and gaseous (including cesium) fission products, coarse porosity, and pellet fracturing A second aspect of the study included the mechanisms of carbon transfer into the steel cladding and its effects on cladding properties In the third aspect, examination of the oxidation of carbide fuels was carried out From this the most efficient head end oxidation treatments have been determined for the reprocessing of mixed carbide fuels

255
High-burnup oxide fuel in European fast reactors. Swanson, K M , Languille, A , Muhling, G *Transactions of the American Nuclear Society (USA),* 60 307-309 (1989) (CONF-891103–)
From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)
The European Collaboration on Fast Reactors is working on the design of a common demonstrate fast reactor, the European Fast Reactor (EFR) designed to be licensable in all the countries of the collaboration The first consistent design of EFR calls for uranium-plutonium oxide fuel assemblies The first core target is a peak burnup of 15 at% at a neutron displacement dose of 135 displacements per atom (dpa) NRT The design targets for later charges are set at 20 at% peak burnup and a 180-dpa NRT displacement dose Achievement of these later burnup targets will reduce fuel cycle costs and increase reactor load factor by allowing a core life of ~6 yr with a refueling cycle length of 2 yr The research and development (R and D) requirement in the core and fuel area is to confirm that these EFR targets can be attained in the preferred pin and subassembly designs using candidate clad and wrapper materials Assessment of the available information by core materials specialists in the collaboration leads to the conclusion that prime contenders for use as pin cladding are cold worked (CW) 15 15-Ti/1 4970 and PE16 for wrapper use, the martensitic steels, FV448/1 4914 and EM10 have been chosen The oxide dispersion strengthened (ODS) ferritics are seen as an alternative option for clad use The final choice for EFR rests on the irradiation performance of subassemblies using these alloys, particularly their ability to consistently reach the EFR targets for 180 to 220 dpa NRT and 20% burnup safely and without failure

256
Experience in mixed-oxide fuel fabrication at the plutonium fuel production facility. Kaneko, Hiromitsu, Shishido, Toshio, Izuhara Shigeomi, Deguchi, Morimoto *Transactions of the American Nuclear Society (USA),* 60 316-317 (1989) (CONF-891103–)
From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit San Francisco, CA (USA) (26-30 Nov 1989)
The Power Reactor and Nuclear Fuel Development Corporation (PNC) of Japan has constructed a new mixed oxide (MOX) fabrication facility, the Plutonium Fuel Production Facility (PFPF) fast breeder reactor (FBR) line at Tokai Works to supply fuel assemblies to the experimental FBR JOYO and the prototype FBR MONJU The construction of PFPF was started in 1982 and all installation was completed in 1987 After testing operations, JOYO fuel production was started in November 1988 The PFPF was constructed as a fully automated MOX fabrication facility by the introduction of a computer-controlled special nuclear material transfer system, which combines automated process equipments This paper describes experiences with fuel fabrication in this line

257
Fuel assembly for LMFBR type reactor. Shirakawa Noriyuki, Endo Hiroshi (to Nippon Atomic Industry Group Co Ltd , Tokyo (Japan) Toshiba Corp Kawasaki, Kanagawa (Japan)) Japan Patent 2-138898/A/ 28 May 1990 Filed date 18 Nov 1988 3p (In Japanese) JAPIO Also available from INPADOC
In a fuel assembly for use in LMFBR type reactors in which fuel pins are contained within a wrapper tube made of a hexagonal stainless steel pipe, if heat removing function of coolants is reduced by inlet blockage accident of a fuel assembly, since the thermal conductivity of the wrapper tube is low, heat-removing effect due to the coolants is small to possibly damage the integrity In view of the above in the present invention, the tubular walls of

the wrapper tube is put between a highly heat conductive thin metal plant such as tungsten from the outside and the inside and the thin plates are bonded by means of high heat conductive rivets made of tungsten As a result, since the thermal conductivity at the inside and the outside of the wrapper tube is improved as compared with the conventional case, the heat removing effect for the fuel pins due to the coolants flowing through the assemblies is increased even if the heat removing function due to the coolants for the fuel assembly is re duced the possibility of imparing the integrity of the fuel pin can be reduced (N H)

258

Fuel exchanger for FBR type reactor. Ozaki Hiroshi (to Fuji Electric Co Ltd Kawasaki Kanagawa (Japan)) Japan Patent 2 129596/A/ 17 May 1990 Filed date 10 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

Crown like oblique protrusions are disposed at the upper and the lower portions of a fuel for positioning to a predetermined direction when the fuel is inserted to a reactor core by a fuel exchanger If this confirmation method is applied to a large scaled reactor, holes are greatly deformed relatively due to the increase of fuel life so that the positioning may sometimes fail Further, since a great number of fuels have to be exchanged at a time, misoperations must be detected Then, there are disposed a orientation rod detachably provided to a fuel catching and releasing gripper, a fuel rotational driving mechanism that can be operated rotationally from the outside of the reactor and a fuel discrimination device comprising a ultrasonic wave tranceiver disposed at a position opposing to a discrimination pattern comprising a combination of circular grooves formed at the top of the fuel In this way, the rotational direction of the fuel can be set and the kind of the reactor core constituent elements caught by the exchanger can be discriminated by the fuel discrimination device in a state where the pantograph arms are folded (N H)

259

Dispersed type reactor. Nakai, Satoru (to Power Reactor and Nuclear Fuel Development Corp , Tokyo (Japan)) Japan Patent 2-128192/A/ 16 May 1990 Filed date 8 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

Since conventional FBR type reactors require sodium pipeways, intermediate heat exchangers and pumps, etc , there is a problem in view of the reliability and the cost Then, according to the present invention, a reactor core and a steam generator of a reduced size are combined as a modular structure and a plurality of them are disposed in a cooling vessel Then the heat generated in the reactor core is transmitted to the steam generator by natural convection and taken out of the reactor In this way sodium pipeways intermediate heat exchang ers and pumps are no more necessary to improve the reliability and economy Change of power can be attained by increasing or decreasing the number of modules Further, since the reactor core is reduced in the size inherent safety is high and even if one module has a trouble heat of the reactor core can be removed by other modules (N H)

260

LMFBR type reactor. Amano, Ken, Goto, Tadashi (to Hitachi Ltd , Tokyo (Japan)) Japan Patent 2-124498/A/ 11 May 1990 Filed date 2 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

In a reactor vessel of a FBR type reactor, especially, a tank type FBR type reactor using liquid metals as coolants, a temperature stratification boundary is formed not only upon occurrence of earthquakes but also upon usual reactor scram to form wave movement having a wavelength substantially corresponding to the diameter of the reactor vessel Then, net like or porous plates are disposed horizontally just above the inlet hole of an intermediate heat exchanger as a portion of gener ating the temperature stratification boundary The plates have an effect of stopping the wave movement rapidly since they effectively absorb the energy of the vertical movement of the temperature stratification boundary and cause fine swirls in the flow upon passage to thereby diffuse energy Further, the fine swirls promote the mixing of high and low temperature sodium to moderate the abrupt temperature gradient at the temperature stratification boundary and to moderate thermal stresses gives to structures In this way, the vertical vibrations of the temperature stratification boundary can be prevented upon reactor scram and thermal fatigues can be prevented without causing periodical thermal stresses to the structures in the vicinity of the temperature stratification boundary (N H)

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FBR reactor. Ota, Shuichi, Endo, Hiroshi (to Nippon Atomic Industry Group Co Ltd , Tokyo (Japan), Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-120692/A/ 8 May 1990 Filed date 31 Oct 1988 3p (In Japanese) JAPIO Also available from INPADOC

In a FBR type reactor, if the flow of coolants is interrupted and fuel pin bundles can no more be cooled, this may lead to a core meltdown accident In view of the above, it has been designed such that the coolants are not boiled and the fuel pin bundles can be cooled sufficiently to ensure the safety on the condition that the rate of closure is less than about 85% even if the coolant inlet closure accident should occur In the present invention, in order to further improve the safety, a high pressure plenum is constituted with at least two independent coolant flow channels from which the coolants flown to a fuel assembly passing through the inlet Accordingly, even if the inlet of the coolant flow channel is clogged by obstacles intruding into one of the coolants flow channels in the high pressure plenum, the rate of closure is kept to less than 50% and the fuel pin bundles can be cooled sufficiently In this way, the safety performance can be improved further as compared with conventional cases (N H)

262

Pump impeller. Madden, M UK Patent 2223538/A/ 11 Apr 1990 6p The Patent Office, Sales Branch, St Mary Cray, Orpington, Kent BR5 3RD

A mixed-flow pump impeller, which may be used, for example, as a primary pump for circulating sodium as the primary coolant in a fast nuclear re actor, is described which comprises an impeller with evenly-spaced blades Some of the blades, which are symmetrically disposed around the axis of rotation of the impeller, extend beyond the ends of the other blades towards the suction side of the pump to form an inducer The channels defined between the extensions of the extended blades follow helical paths parallel to the axis of rotation The leading edges of the unextended blades are interposed between the extended blades in the region of divergence of flow from the axis of rotation The provision of the inducer reduces the risk of cavitation in the pump which could cause rapid wear of the impeller A shroud may be provided for the unextended blades (author)

Auxiliary, Mobile, Package, and Transportable

263
(EGG M-90309)
Unique features of space reactors. Buden, D (EG and G Idaho, Inc, Idaho Falls, ID (USA)) [1990] Contract AC07-76ID01570 7p (CONF-900917-24) NTIS, PC A02/MF A01 - OSTI, GPO Dep Order Number DE91001910

From American Nuclear Society topical meeting on safety of non-commercial nuclear reactor research and irradiation facilities, Boise, ID (USA) (30 Sep - 3 oct 1990)

Space reactors are designed to meet a unique set of requirements, they must be sufficiently compact to be launched in a rocket to their operational location operate for many years without maintenance and servicing operate in extreme environments and reject heat by radiation to space To meet these restrictions operating temperatures are much greater than in terrestrial power plants and the reactors tend to have a fast neutron spectrum Currently a new generation of space reactor power plants is being developed The major effort is in the SP-100 program, where the power plant is being designed for seven years of full power, and no maintenance operation at a reactor outlet operating temperature of 1350 K 8 refs 3 figs, 1 tab

264
For Europe the difficult question of space nuclear power systems. Poher, C *Aeronautique et l'Astronautique (Paris) (France)*, No 142, 32-35 (Mar 1990) (In French)

The paper estimates the feasibility of such systems in Europe, their cost and best development procedures, and propose some choices for different applications Good candidates for the use of nuclear power systems in space after 2005 are studied (1) an electric OTV dimensioned for commercial launches in GEO by Ariane 5 of very heavy payloads, obtaining results about the mass traffic needed to obtain an economically interesting system (2) Space based radars (3) A small amount of activity have taken into account problems concerning space stations, manned, visited or unmanned (micro-gravity factories) (4) Planetary manned exploration missions are examined mainly towards Mars For each mission has been studied an optimized nulcear space generator, and its performances have been compared with concurrent technologies Nuclear generators, always appearing first the best choice, were finally rejected for secondary reasons having nothing to do with safety, performance and radiations The nuclear generators were only rejected for the time where a reasonable future can be predicted

265
Reliability and mass analysis of lunar-based reactor/stirling cycle power plants. Bloomfield, H S *Transactions of the American Nuclear Society (USA)*, 60 372 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco CA (USA) (26-30 Nov 1989)

The purpose of this analysis was to determine the mass and reliability characteristics of nuclear reactor/Stirling cycle power plant configurations that could provide 800 kW of electricity for a proposed National Aeronautics and Space Administration (NASA) lunar surface base The specific goal of the work was to define and characterize minimum mass power plant configurations that could provide an acceptable system reliability risk A generic power plant design concept that exhibited potential construction feasibility and met human-rated radiation dose criteria was developed to serve as the basis for the power plant configurations assessed in this study A combinatorial reliability analysis model based on parallel, redundant, series, and r-out-of-n system and component configurations was used to improve system reliability to an acceptable risk level As a result of this study, an increased awareness of the importance of reliability analyses on high-capacity space power system design configurations has evolved, and future NASA mission application studies requiring high power levels for electric propulsion and orbital or planetary surface operations will benefit from this type of analysis

266
Reactor power for space exploration. Schnyer, A D *Transactions of the American Nuclear Society (USA)*, 60 472-473 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Potential 21st century missions envisioned by National Aeronautics and Space Administration (NASA) planners encompass ambitious, wide-ranging human and robotic solar system exploration objectives and scenarios A critical common element in many of these future civil space mission initiatives is the ability to generate, with a very high degree of reliability, the considerable amounts of power needed to realize the mission goals The extended duration and/or high power level requirements for many missions and, in instances, the lack of adequate solar energy flux for others, render the use of versatile nuclear power sources as either missions-enabling or very advantageous Further, the use of high-performance reactor systems, when coupled with very high impulse electric propulsion systems, can enable or significantly enhance both human near planets operations and robotic scientific missions to the very farthest reaches of the solar system It is important that this nation continue to develop the means of acquiring a space reactor power source to ensure availability at such time that approved missions and possibly political considerations warrant its use

267
SP-100 program - Where we are today. Bailey, H S *Transactions of the American Nuclear Society (USA)*, 60 474-475 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The SP-100 ground engineering system (GES) program is in the process of validating the technology/components on which the generic flight system (GFS) design is based A major step in this validation process is the nuclear assembly test (NAT) to be built by General Electric and operated by Westinghouse Hanford Company in an existing containment building at the US Department of Energy's Hanford facilities in the state of Washington Other segments of the GES program include the development of the thermoelectric converter assembly, the thermoelectromagnetic (TEM) pump, and the gas separator The evolution of all these segments from design to hardware is in full swing Nuclear fuel development and manufacture is under way at Los Alamos National Laboratory in New Mexico An important activity in the GES program is materials development and testing Oak Ridge National Laboratory (ORNL) plays a key role in this effort with support from Westinghouse Advanced Energy Systems Encouraging progress in other GES activities not directly associated with the NAT includes the successful manufacture of multicouple thermoelectric modules incorporating the compliant pad coupling to the heat source and sink and the necessary sapphire electrical insulation in a mass efficient configuration Successful tests of both the TEM pump and gas separator concepts lend validity to proposed flight system designs

268
SP-100 ground engineering system nuclear assembly test. Kruger, G B , Mahaffey, M K *Transactions of the American Nuclear Society (USA),* 60 475-477 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit San Francisco CA (USA) (26-30 Nov 1989)

SP-100 is a space power system that is being developed by the General Electric Company to meet future space electrical power requirements The ground testing of an SP-100 prototypical reactor system will be conducted at the Westinghouse Hanford Company site located at Richland Washington Westinghouse Hanford will design and supply all associated site facility support systems General Electric will supply the nuclear assembly test (NAT) to the site The goal of the reactor ground test system is to establish confidence in the design maturity of the SP-100 space reactor power system and resolve the technical issues necessary for the development of a light mission design This paper presents a brief overview of the SP-100 GES NAT Results of the test program will provide substantial data that will be used to validate the flight system design in preparation for a future flight mission

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SP-100 from ground demonstration to flight validation. Buden, D *Transactions of the American Nuclear Society (USA),* 60 473-474 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The SP-100 program is in the midst of developing and demonstrating the technology of a liquid-metal-cooled fast reactor using thermoelectric thermal to-electric conversion devices for space power applications in the range of tens to hundreds of kilowatts The current ground engineering system (GES) design and development phase will demonstrate the readiness of the technology building blocks and the system to proceed to flight system validation This phase includes the demonstration of a 2 4-MW(thermal) reactor in the nuclear assembly test (NAT) and aerospace subsystem in the integrated assembly test (IAT) The next phase in the SP-100 development, now being planned, is to be a flight demonstration of the readiness of the technology to be incorporated into future military and civilian missions This planning will answer questions concerning the logical progression of the GES to the flight validation experiment Important issues in planning the orderly transition include answering the need to plan for a second reactor ground test, the method to be used to test the SP-100 for acceptance for flight, the need for the IAT prior to the flight-test configuration design, the efficient use of facilities for GES and the flight experiment, and whether the NAT should be modified based on flight experiment planning

270
SP-100, a flexible technology for space power. Smith, M A , Stephen, J D , Stewart, S L , Cho, Hwang, Kirpich A , Shepard, N F *Transactions of the American Nuclear Society (USA),* 60 474 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Power requirements for the ambitious space missions of the future exceed the limits of current power system availability Space reactor power meets this need The US is developing space reactor power system technology to meet the projected civilian and defense needs for the 1990s and beyond This program, known as SP-100, will remove the restrictions on electrical power generation that have limited space missions and will enable the fuller exploration and utilization of space The SP-100 program is developing technology to provide tens of hundreds of kilowatts of reliable, long-life space electrical power The program is sponsored by the Department of Defense, the Department of Energy, and the National Aeronautics and Space Administration (NASA) To focus the SP-100 program technology development for space nuclear power, a generic flight system (GFS) has been designed to meet a broad set of requirements characteristic of a variety of potential earth orbiting or interplanetary missions The net power output selected is 100 kW(electric) No new technology is required for specific mission configurations at lower or higher electrical power output based on this generic system

271
Component design challenges for the ground-based SP-100 nuclear assembly test. Markley, R A , Disney, R K , Brown, G B *Transactions of the American Nuclear Society (USA),* 60 477-478 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The SP-100 ground engineering system (GES) program involves a ground test of the nuclear subsystems to demonstrate their design The GES nuclear assembly test (NAT) will be performed in a simulated space environment within a vessel maintained at ultrahigh vacuum The NAT employs a radiation shielding system that is comprised of both prototypical and nonprototypical shield subsystems to attenuate the reactor radiation leakage and also nonprototypical heat transport subsystems to remove the heat generated by the reactor The reactor is cooled by liquid lithium, which will operate at temperatures prototypical of the flight system In designing the components for these systems, a number of design challenges were encountered in meeting the operational requirements of the simulated space environment (and where necessary, prototypical requirements) while also accommodating the restrictions of a ground-based test facility with its limited available space This paper presents a discussion of the design challenges associated with the radiation shield subsystem components and key components of the heat transport systems

RESEARCH, TEST AND EXPERIMENTAL REACTORS

272
(EGG-CS-8668)
Neutron spectrum studies in the ATR center lobe positions H-02, H-10 and H-14. Rogers, J W , Anderl, R A (EG and G Idaho Inc , Idaho Falls ID (USA)) Sep 1989 Contract AC07-76ID01570 105p NTIS PC A06/MF A01 OSTI, INIS, GPO Dep Order Number DE91001925

Neutron spectrum studies have been conducted for the ATR center lobe positions H-02, H-10 and H-14 Measurements were obtained with neutron activation monitors which have responses in the energy range between "cadmium cutoff" and about 13 MeV Calculated neutron energy spectra were obtained from reactor models and transport/diffusion theory computations Measured and calculated reaction rates were processed with least-squares-data-adjustment theory codes to produce measurements/adjusted neutron spectra for each position Spectral averaged cross sections and response functions for each reaction are presented for typical H-hole spectra Displacements-per-atom damage cross sections for Fe, Cr and Ni have been calculated for representative H-hole spectra Comparisons are made between the measurements adjusted neutron spectra and a pure fission neutron spectrum (Watt) These studies were performed to provide information related to neutron monitoring, neutronics calculations and irradiation experiments in the reactor 10 refs , 19 figs , 18 tabs

273
(EGG M-89365)
Pressure Fed Nuclear Thermal Rockets for space missions. Leyse, C F Madsen W W , Ramsthaler, J H , Schnitzler, B G (EG and G Idaho, Inc , Idaho Falls, ID (USA)) Aug 1989 Contract AC07-76ID01570 9p (CONF-900109-26) NTIS, PC A02/MF A01 - OSTI, GPO Dep Order Number DE91001847

From 7 symposium on space nuclear power systems, Albuquerque, NM (USA) (7-11 Jan 1990)

The National Space Policy includes a long range goal of expanding human presence and activity beyond Earth orbit into the solar system This has renewed interest in the potential application of Nuclear Thermal Rockets (NTR) to space flight, particularly for human expeditions to the Moon and Mars Recent NASA studies consider applications of the previously developed NERVA (Nuclear Engine for Rocket Vehicle Application) technology and the more advanced gas core reactors and show their potential advantages in reducing the initial mass in Earth orbit (IMEO) compared to advanced chemical rocket engines Application of NERVA technology will require reestablishing the prior technological base or extending it to an advanced NERVA type engine, while the gas core NTR will require an extensive high risk research and development program A technology intermediate between NERVA and the gas core NTR is a low pressure engine based on solid fuel, a Pressure Fed NTR (PFNTR) In addition to the simplicity of the gas pressurized engine cycle, the PFNTR takes advantage of the dissociation of hydrogen-the increases in specific impulse become significant as the chamber pressure decreases below 1 0 MPa (10 atmospheres) and the chamber temperature increases above 3000 K The developmental status of technology applicable to a Pressure Fed Nuclear Thermal Rocket (PFNTR) lies between that of the NERVA engine and the gas core NTR (GCNTR) This document investigates PFNTR performance and provides typical mission analyses

274
(EGG-M--89450)
RELAP5 based engineering simulator. Charlton, T R , Laats, E T , Burtt, J D (EG and G Idaho, Inc , Idaho Falls, ID (USA)) [1990] Contract AC07-76ID01570 14p (CONF-900494-7) NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91001934

From 1990 SCS Eastern multiconference, Nashville, TN (USA) (23-29 Apr 1990)

The INEL Engineering Simulation Center was established in 1988 to provide a modern, flexible, state of-the-art simulation facility This facility and two of the major projects which are part of the simulation center, the Advance Test Reactor (ATR) engineering simulator project and the Experimental Breeder Reactor II (EBR-II) advanced reactor control system, have been the subject of several papers in the past few years Two components of the ATR engineering simulator project, RELAP5 and the Nuclear Plant Analyzer (NPA), have recently been improved significantly This paper will present an overview of the INEL Engineering Simulation Center, and discuss the RELAP5/MOD3 and NPA/MOD1 codes, specifically how they are being used at the INEL Engineering Simulation Center It will provide an update on the modifications to these two codes and their application to the ATR engineering simulator project, as well as, a discussion on the reactor system representation, control system modeling, two phase flow and heat transfer modeling It will also discuss how these two codes are providing desktop, stand-alone reactor simulation 12 refs , 2 figs

275
(EGG-M-90012)
A methodology for existing system upgrade to current ASME standards and system lifetime extension. Burr, T K , Dwight, J E Jr , Hawkes, G L , Pace, N E (EG and G Idaho, Inc , Idaho Falls, ID (USA)) Jan 1990 Contract AC07-76ID01570 5p (CONF-900917-25) NTIS, PC A01/MF A01, OSTI, INIS, GPO Dep Order Number DE91001996

From American Nuclear Society topical meeting on safety of non-commercial nuclear reactor research and irradiation facilities, Boise, ID (USA) (30 Sep - 3 oct 1990)

In the wake of the Chernobyl events, there has been an increase in the awareness and review of government operated reactors both internationally, and within the United States Government reactors have recently come under increased and indepth scrutiny Department of Energy Secretary Hodel committed to a review of the safety of non-commercial reactors and irradiation facilities within the department The increased attention has been in the areas of accident response, PRA of the facilities, environmental impacts, and the construction and associated standards for the facilities This paper focuses on the system qualifications to current standards Specifically, this paper discusses a method used for upgrading an existing high pressure nuclear system to current ASME Code standards and to extend the system's lifetime This paper reports the methods used in an attempt to qualify components of the Advanced Test Reactor (ATR) located at the Idaho National Engineering Laboratory (INEL) to current ASME Code Section III standards

276
(EGG-M-90054)
A structured approach to evaluating aging of the advanced test reactor. Dwight, J E (EG and G Idaho, Inc , Idaho Falls, ID (USA)) Jan 1990 Contract AC07-76ID01570 16p (CONF-900917-21) NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91001937

From American Nuclear Society topical meeting on safety of noncommercial nuclear reactor research and irradiation facilities, Boise, ID (USA) (30 Sep - 3 oct 1990)

An aging evaluation program has been developed for the United States Department of Energy s Advanced Test Reactor to support the current goal of operation through the year 2014 and beyond The Aging Evaluation and Life Extension Program (AELEX) employs a three phased approach In Phases 1 and 2, now complete, components were identified, categorized and prioritized Critical components were selected and aging mechanisms for the critical components identified An initial evaluation of the critical components was performed and extended life operation for the plant appears to be both technically and economically feasible Detailed evaluations of the critical components are now in progress in the early stages of Phase 3 Some results are available Evaluations of many noncritical components and refinements to the program based on probabilistic risk assessment results will follow in later stages of Phase 3 6 refs 2 figs , 5 tabs

277
(ITEF-82-89)
Lithium converter of reactor neutrinos in antineutrino. Lyutostanskij, Yu S Lyashuk, V I (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow (USSR) Inst Teoreticheskoj i Ehksperimental'noj Fiziki) 1989 27p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91610203

The questions of developing lithium converter of the reactor neutrons in an tineutrino operating at dynamic regime in the scheme with the cycle circulation of the high purified lithium (by ^{7}Li isotope) through the converter are considered The scheme allows to localize the ^{8}Li β-decay ($T_{1/2}=0.84$ s) in the reservoir near the detector and so to design the hard spectrum lithium ν-tilde$_e$-source ($E_{max}\simeq 13$ MeV) at the distance from the active zone being the soft-spectrum ν tilde$_e$-source The expressions for the lithium ν-tilde$_e$ flux from the converter, reservoir and conveyance channel are obtained 9 refs 8 figs 1 tab

278
(JAERI-M-90 112)
Measurement and calculations of neutron interaction effects of a two-coupled system in water. Miyoshi, Yoshinori, Suzaki, Takenori, Ishikawa, Toshimitsu, Kobayashi, Iwao (Japan Atomic Energy Research Inst , Tokyo (Japan)) Jul 1990 52p NTIS (US Sales Only), PC A04/MF A01 Order Number DE91723322

A critical experiment on the interaction effect between two rectangular cores moderated with light water was performed at Tank-type Critical Assembly (TCA) of JAERI Each core was composed of the array of UO_2 fuel rods of which ^{235}U enrichment was 2 6 w/o The lattice cell was a square type of which the pitch was 1 956 cm corresponding to the water-to fuel volume ratio of 1 83 From the obtained critical water heights, reactivity effects from one unit to another through water gap and negative reactivity of water gap and subcriticality of single units were measured by use of water level worth method as a function of the distance between two units Calculations for critical configurations were also made with Monte Carlo code KENO-IV in JACS system, and compared with the measured reactivity effects (author)

279
(JAERI-M-90-115)
Integrity assessment of test fuel assemblies of the High Temperature Engineering Test Reactor. Hayashi, Kimio, Shiozawa, Shusaku, Fukuda, Kousaku (Japan Atomic Energy Research Inst , Tokyo (Japan)) Jul 1990 83p NTIS (US Sales Only), PC A05/MF A01 Order Number DE91723357

Assessment of integrity has been made on the B-type fuel assemblies, which will be loaded in the High Temperature Engineering Test Reactor (HTTR) as test fuel assemblies Specifications of coated fuel particles for the B-1 type fuel assembly have been slightly changed in the fuel kernel diameter and thickness of coating layers from those for the A-type fuel assembly, which is employed as the driver fuel These changes have been directed toward safer side in developing this advanced fuel for use up to higher burnups at higher temperatures The B-2 type fuel assembly uses the zirconium carbide (ZrC) coating layer with excellent high-temperature chemical stability, instead of the silicon carbide (SiC) layer This change has lead to demonstration of its better performance than the A-type fuel assembly in the kernel migration, corrosion by fission products including palladium, and coating failure at extremely high temperatures The B-3 type fuel as sembly adopts the $(U,Th)O_2$ kernel - SiC TRISO coated fuel articles The service condition (1000degC and 22,000 MWd/t) of the B-3 type fuel assembly is decided as the range within which the performance data of the fuel have been sufficiently obtained Thus, it has been judged that the integrity of these B-type fuel assemblies will be maintained under the normal operating conditions of the HTTR Moreover, the validity of the permissible design limit of the fuel has been confirmed, which requires that the fuel temperature shall not exceed 1,600degC at anticipated operational transients (author)

280
(JAERI-M-90-118)
Experimental test results of multichannel test rig of T_1 test section, 4. Takase, Kazuyuki, Hino, Ryutaro, Miyamoto, Yoshiaki (Japan Atomic Energy Research Inst , Tokyo (Japan)) Aug 1990 37p NTIS (US Sales Only), PC A03/MF A01 Order Number DE91723358

The multi-channel test rig (T_{1-M}) of the fuel stack test section in the helium engineering demonstration loop (HENDEL) is a large-scale experimental facility which simulates one fuel column of the HTTR core A crossflow test was carried out using the T_{1-M} The objectives of this test are to investigate the thermal and hydraulic characteristics in the fuel stack under the conditions that produced the crossflow through a gap from the outside of the graphite blocks into twelve coolant channels the crossflow was forcibly produced by a parallel gap situated between the third and fourth blocks from the top of those in the heated section mounted in the vertical direction Crossflow rate were about a half of the total flow rate of helium gas in T_{1-M} for heated flow A decrease of the gap width raised the crossflow rates in outer channels (No 7≈12) and also reduced those in inner channels (No 1≈6) It was found that the crossflow affected flow redistributions in the channels and temperature distributions of the fuel rods (author)

281
(K/CSD/TM-79)
Calculations for HFIR [High Flux Isotope Reactor] fuel plate non-bonding and fuel segregation uncertainty factors. Kirkpatrick, J R (Oak Ridge Gaseous Diffusion Plant, TN (USA)) Oct 1990 Contract AC05 84OT21400 41p NTIS PC A03/MF A01 OSTI INIS, GPO Dep Order Number DE91001086

The effects of non-bonds and of fuel segregation on the package factors of the heat flux in the High Flux Isotope Reactor (HFIR) are examined The effects of the two defects are examined both separately and together It is concluded that the peaking factors that are used in the present HFIR thermal analysis code are conservative and thus no changes in the peaking factors are necessary to continue to ensure that HFIR is safe A study was made of the effect of the non-bond spot diameter on the peaking factor The conclusion is that the spot can have diameter more than three times the maximum value allowed by the specifications before the peaking factor is greater than the maximum value specified in the present HFIR thermal analysis code 6 refs , 7 figs , 8 tabs

282
(ORNL/M-1317)
Advanced Neutron Source equipment data base. Coffin, D B (Oak Ridge National Lab , TN (USA)) Aug 1990 Contract AC05-84OR21400 21p NTIS, PC A03/MF A01 - OSTI, GPO Dep Order Number DE91004052

The Advanced Neutron Source (ANS) is a new experimental facility planned to meet the national need for an intense steady-state source of neutrons It will be open for use by scientists from universities, industry and other federal laboratories The ANS will be equipped with an initial complement of advanced instruments for neutron scattering and nuclear physics research, with facilities for isotope production and for the study of materials in high radiation fields The central structure is a 60-m (~200-ft) diam cylindrical, domed reactor building This building will house the reactor itself, with its lower floors dedicated to beam and irradiation experiments and with a high-bay floor dedicated to reactor operations A reactor support building, to be adjacent to the reactor building, will house other large reactor equipment and the general support equipment not located in the reactor building The primary heat exchanger and circulating pumps will be located in cell banks within reactor containment The guide hall building, connected to the reactor dome outside reactor containment, is dedicated to beam experiment use The fourth building will be an office building serving both the extensive user community and the reactor operations staff These buildings will contain many of the systems needed for operation of the ANS and will be comprised of equipment requiring specification of performance test and operating parameters The number of equipment items the possibility for multiple application of a particular piece of equipment, and the need for a single source of information for all equipment led to a requirement to develop and equipment-related data base 3 refs 2 figs , 1 tab

283
(ORNL/TM-11676)
Bulk shielding facility semi-annual report, January–June 1990. Laughlin, D L , Coleman, G H (Oak Ridge National Lab , TN (USA)) Nov 1990 Contract AC05-84OR21400 17p NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91002931

The Bulk Shielding Reactor (BSR) remained shut down during January, February, March, April, May, and June Water-quality control in both the reactor primary and secondary cooling systems was satisfactory Maintenance and changes are described The Pool Critical Assembly (PCA) remains shut down, but surveillance tests are described

284
Physics studies of higher actinide consumption in an LMR. Hill, R N , Wade, D C , Fujita, E K , Khalil, H (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp I 83-I 96 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

The core physics aspects of the transuranic burning potential of the integral Fast Reactor (IFR) are assessed The actinide behavior in fissile self-sufficient IFR closed cycles of 1200 MWt size is characterized, and the transuranic isotopics and risk potential of the working inventory are compared to those from a once-through LWR The core neutronic performance effects of rare earth impurities present in the recycled fuel are addressed Fuel cycle strategies for burning transuranics from an external source are discussed, and specialized actinide burner designs are described

285
The measurement of the effective delayed-neutron fraction in the fast critical assembly BFS with uranium-plutonium metal fuel. Soule, R , Bertrand, P , Pont, J , Gauthier, J C , Granget, G , Avramov, A M , Doulin, V A , Zhuravlev, V I , Kochetkov, A L , Momontov, V F (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp III 95-III 106 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Precise measurements of the effective delayed neutron fraction β_{eff} in zero power critical assemblies close in composition to fast power reactor geometries have been of interest for some time Measurements with uncertainties of ≤5% would allow appreciable refinement of the absolute scale of reactivity In the BFS-facility for example, discrepancies in excess of 10% between experimental and calculated values of the central coefficients of reactivity of fuel are not infrequent In the framework of franco-soviet collaboration, measurements of β_{eff} (EROS experiment) were performed in Spring '89 in the BFS facility Two different techniques were used The first, based on the well known methods of determination of the californium source pseudo-reactivity, was carried out by the soviet team The second technique, based on reactor noise measurements was carried out by the French team This technique was first used during the BALZAC program in the MASURCA critical facility and since the method is still under developement, the associated results must accordingly be considered as preliminary All the measurements were performed on the one zone assembly BFS 55-LA close in composition to the plutonium breeder with metallic fuel and sodium coolant

286
Data uncertainty reduction in high converter reactor designs using PROTEUS phase II integral experiments. White, J R , Delorey, T F (Societe Francaise d Energie Nucleaire (SFEN), 75 - Paris (France)) pp III 107-III 116 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

A detailed sensitivity and uncertainty analysis was performed for several parameters of interest in the design of the high conversion reactor (HCR) concept The main goals of this work were to determine the response standard deviation due to basic nuclear data uncertainties and to incorporate integral experiment information from the PROTEUS facility to reduce the computed uncertainties wherever possible This paper highlights the results for K and five important reaction rate ratios that were part of the measurement program in the PROTEUS phase II experiments The computed correlation coefficients between the PROTEUS and HCR models were uniformly high which indicates that a considerable reduction in uncertainty can be achieved (within measurement uncertainties)

287
A burnable poison pin benchmark study in Dimple. Knipe, A D , Burbidge, B L H , Franklin, B M (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp IV 1-IV 12 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Studies in the DIMPLE low power reactor are providing benchmark experimental data for the validation of the calculational methods employed in the design and operation of thermal power reactors In the most recent phase of the programme a series of burnable poison arrays were included in a cruciform lattice of 3% enriched uranium dioxide fuel pins Extensive reaction-rate distribution measurements coupled with diagnostic fine structure measurements through a fuel pin, poison pin and simulated empty guide thimble, have provided detailed data for the assessment of the accuracy of pin power predictions

288
Evaluation of absorber experiments in the assemblies SNEAK 12A and SNEAK 12C2. Henneges, G , Helm, F (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp V 51-V 62 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation Paris (France) (23-26 Apr 1990)

The reactivity effects of absorbers of different enrichment and geometry as well as of different arrangements were measured in the critical assemblies SNEAK 12A a single-zone core, fueled with enriched uranium metal plates and SNEAK 12C2 a core which had a central test zone fueled with PuO_2UO_2 plates surrounded by a buffer and a driver zone Additionally substitution of B_4C filled rodlets of different ^{10}B-enrichment by $ZrH_{1.8}$ was studied in SNEAK 12C2 The reactivity effects were calculated using current Kernforschungszentrum Karlsruhe methods and data The evaluation yields similar results for both assemblies For most cases investigated, satisfactory agreement between theory and experiment is reached when two-dimensional transport eigen-value calculations and XYZ-diffusion methods are combined, together with a variety of more sophisticated corrections Finally some typical experimental and calculated reactivities of both assemblies are compared and the results found discussed

PRODUCTION REACTORS

289
(EGG-M-90343)
Simulation of three-dimensional hydrodynamic components with a one-dimensional transient analysis code. Shaw, R A , Davis, C B (EG and G Idaho, Inc , Idaho Falls, ID (USA)) [1990] Contract AC07-76ID01570 5p (CONF-901154-1) NTIS, PC A01/MF A01 - OSTI, GPO Dep Order Number DE91001816

From Meeting of the Fluid Dynamics Division of the American Physical Society, Ithaca, NY (USA) (18-20 Nov 1990)

Significant multidimensional pressure gradients occur in the water plenum region of the reactors at the Savannah River Site (SRS) A multidimensional RELAP5 input model of the L-Reactor was developed and benchmarked against SRS data Although RELAP5 is a one-dimensional code, its cross-flow junction allows a multidimensional analysis capability RELAP5 and the model of L-Reactor calculated water plenum pressures that were in good agreement with measured values for tests with symmetric and asymmetric flow patterns within the plenum The results indicate that a one-dimensional code such as RELAP5, in conjunction with a carefully designed input model, can be used to predict hydraulic response even when large multidimensional effects are present 6 refs , 6 figs

290
Higher actinides transmutation using higher actinide burner reactors. Mukaiyama T , Takano, H , Takizuka, T , Ogawa, T , Gunji, Y , Okajima, S (Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris (France)) pp I 97-I 107 of The physics of reactors operation, design and computation Volume 1 Paris (FR), Societe Francaise pour l'Energie Nucleaire (1990) 593p (CONF-900409-)

From International conference on the physics of reactors operation, design and computation, Paris (France) (23-26 Apr 1990)

Nuclear transmutation of long-lived nuclides into shorter-lived nuclides is the attractive option which may answer the high-level radioactive waste management One of the techniques of higher actinides transmutation is to recycle these nuclides into fast reactors

In this paper, designing of a higher actinides burner reactor and fuel cycle facilities is discussed to assess the concept of these reactors The characteristics of transmutation in burner reactors is compared with those in power reactors

THEORY AND CALCULATION

291
(CTA IEAv NT-017/89)
Introduction of gadolinium in the library of Leopard code. Claro L H Menezes A (Centro Tecnico Aeroespacial (CTA IEAv) Sao Jose dos Campos SP (Brazil) Inst de Estudos Avancados) Dec 1989 15p NTIS (US Sales Only) PC A03/MF A01 OSTI INIS Order Number DE91607631

The materials Gd-154 Gd 155, Gd 156 and Gd 157 were included in the LEOPARD code library at the request of FURNAS Centrais Eletricas S A Results from comparison of LEOPARD and WIMSD/4 codes for a typical cell with 7 burnup steps, are presented (author)

292
(LA-UR-90-3603)
Three dimensional transport benchmark exercise using THREEDANT. Alcouffe, R E (Los Alamos National Lab , NM (USA)) [1990] Contract W-7405-ENG-36 15p (CONF-9010229-1) NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91001861

From Specialist s meeting on three-dimensional neutron transport benchmarks Paris (France) (22 23 Oct 1990)

As part of the effort to assess the ability to perform three dimensional transport calculations to solve problems in reactor physics we describe the THREEDANT code and its application to the set of three dimensional benchmark problems proposed by Prof T Takeda As part of this bench marking activity we display some key indicators as to computational performance and efficiency while displaying the sensitivity of the eigenvalue to Sn order and to spatial mesh size in each of the problems In order to understand what is being displayed, we summarize the solution strategy incorporated in the code 1 ref , 3 figs

293
(WSRC-TR-90-189)
Benchmarking report for WIGGLE: A one-dimensional transient diffusion theory code. Pevey, R E (Westinghouse Savannah River Co , Aiken, SC (USA)) Nov 1990 Contract AC09-89SR18035 19p Distribution UC-706 NTIS, PC A03/MF A01 - OSTI, GPO Dep Order Number DE91004119

WIGGLE is a static/transient one-dimensional diffusion theory calculation written to estimate the axial power profile while safety rods are falling during a scram The code is used in the LOCA Limits Analysis Package (LLAP), a part of the SRS system for calculating thermal-hydraulic limits Since WIGGLE was designed to be implemented through LLAP and not as a stand-alone code it consists entirely of subroutines the problem data must be passed to it from a driver routine This project concerned the verification of WIGGLE which limited it to the determination that WIGGLE is correctly implementing the transient 1D diffusion equation The approach was to compare the results of the code with three analytic solutions a static homogeneous calculation of the pre-accident power profile (without end-fittings), a static heterogeneous calculation of the pre-accident power profile (includes end-fittings), and a transient calculation designed to test the time-dependent calculational ability The results of all three calculations were essentially identical to the analytical solutions, thus giving us confidence that WIGGLE is correctly solving the one-dimensional time-dependent diffusion equation

294
Nuclear systems 2. Todreas, N E , Kazimi M J New York, NY (USA), Hemisphere Publishing (1990) 506p Hemisphere Publishing 79 Madison Avenue, New York NY 10016 (USA)

The book covers elements of thermal hydraulic design and analysis of the core of nuclear reactor Other components of the nuclear power plant such as the pressurizer, the containment and the entire primary coolant system are addressed The book reflects the importance of such considerations in thermal engineering of a modern nuclear power plant

295
Nuclear systems 1. Todreas, N E , Kazimi, M J New York, NY (USA), Hemisphere Publishing (1990) 705p Hemisphere Publishing, 79 Madison Avenue, New York, NY 10016 (USA)

The book covers thermal hydraulic design fundamentals and analysis of the core of a nuclear reactor Other components of the nuclear power plant, such as the pressurizer, the containment and the entire primary coolant system are addressed The book reflects the importance of such considerations in thermal engineering of a modern nuclear power plant

296
Simple analytical fits to fission-product decay heat. Sridharan, M S *Annals of Nuclear Energy (UK),* 17 No 7, 389-392 (1990)

Several computer codes exist for detailed summation calculation of fission product decay heat The results of this calculation are usually fitted to an analytical function This fit is useful when extensive routine estimates are re quired to be made Some analytical fits involving many exponential terms, have been reported in literature In this paper it is shown that another type of fit, simple in nature, can be obtained and employed for decay heat prediction This is accomplished by dividing the time range into a number of segments Three case studies involving 47/14/1 time segments are undertaken and analytical fits corresponding to these segments are obtained The fitting accuracy relative to the summation calculation is examined The overall deviation (for the irradiation and cooling time ranges specified in this paper) is found to be ± 1% for the 47 time segments case, ± 6% for the 14 time segments case and ± 20% for a single time segment case These results appear to be satisfactory The work carried out here is confined to fast fissions in ^{239}Pu (author)

297
Discrete ordinates methods for radiation transport. Badruzzaman, A , Fan, W C *Transactions of the American Nuclear Society (USA),* 60 332 335 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The discrete ordinates (S_N) method, first developed for stellar atmospheres, has been used extensively on various other radiation transport problems In reactor analysis the method is generally used to generate parameters for design models based on more approximate but less expensive methods (such as diffusion theory) so that the spatial spectrum coupling is represented accurately on a microscopic reaction rate level It has a decisive advantage over Monte Carlo methods in computing the pin and assembly power profiles In shielding problems where the penetration of the radiation can be deep, the method is used widely in design calculations In oil-well logging problems, which also involve deep penetration and have a stringent accuracy requirement on the detector responses, the method complements the Monte Carlo techniques One early application of the S_N method was on one dimensional radiative transfer problems The discrete ordinates method has also been used in charged-particle transport problems While the method has been applied primarily to static problems, one-dimensional time-dependent codes have existed since the early 1970s In this paper the authors briefly review the basic method illustrate its applications, discuss its merits and pitfalls, and enumerate the recent advances in the attendant numerical techniques that have enhanced the capabilities of the method

298

Commercial exploitation of finite element codes for neutron transport. Ackroyd R T , Issa, J G *Transactions of the American Nuclear Society (USA),* 60 335 336 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco CA (USA) (26-30 Nov 1989)

The finite element method for neutron transport is in commercial use in the United Kingdom for shielding and criticality problems It is also applied to time dependent problems arising in the field of oil-well logging The present capability of the method is described, and the developments in hand to accelerate calculations are outlined The codes that have been developed in the United Kingdom are FELTRAN, TRIPAC, and MARC Complex shapes can be modeled with finite elements as realistically as in Monte Carlo calculations Thus the finite element and Monte Carlo methods provide an independent means of cross checking a safety assessment for finalized design The finite element method is particularly useful for scoping design studies, because it gives the angular fluxes everywhere in the system These fluxes provide guidance on the efficacy of the design features

299

Monte Carlo methods for particle transport. Martin, W R , Brown, F B *Transactions of the American Nuclear Society (USA),* 60 336-337 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco CA (USA) (26-30 Nov 1989)

Monte Carlo codes for simulating particle transport were among the first applications of computers in the 1940s Since then, advances in computer hardware, software, Monte Carlo methodology, and cross-section data availability have led to the development of a large number of general-purpose Monte Carlo production codes Today these codes are used extensively to simulate particle transport in applications such as reactor core analysis, radiation shielding, particle detector design and analysis, oil-well logging, high-energy particle physics simulation etc A partial list of general-purpose production codes includes MCNP, MORSE KENO, TART, MONK6, TRIPOLI, SAM CE, RCP, PACER 05R, RACER, GEANT, and McBEND There have been a number of recent review papers on Monte Carlo methods, including one that focused on variance reduction techniques and one that summarized the status of vectorized Monte Carlo Therefore, this paper addresses several broader topics What capabilities are common to typical production-level Monte Carlo codes? Why are there so many production codes? What are current areas of interest for Monte Carlo methods development? What is the impact of advanced computer architectures on Monte Carlo methods?

300

Collision probability methods - An update. Sanchez, R *Transactions of the American Nuclear Society (USA),* 60 338-339 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

A preliminary step in reactor coarse-mesh calculations is the preparation of an adequate macroscopic cross section library by spatial smearing and group collapsing of detailed cell and/or assembly multigroup transport calculations Currently, only two different techniques are regularly used for the determination of the multigroup transport flux in heterogeneous cells or assemblies (1) methods based on the discretization of the integrodifferential transport equation and (2) collision probability methods for the numerical solution of the integral transport equation Collision probability (CP) methods are derived either from the direct use of a quadrature formula or by the expansion of the unknown scalar flux in terms of representation functions The new computer generation, with faster processing and a larger central memory that can support precise tabulations of transcendental functions, has considerably reduced the cost of multidimensional CP calculations CP methods have come of age and are currently used in bidimensional routine reactor calculations through the world Recent developments in CP methods include double-heterogeneity approximations and multilevel methods Current CP research in France is oriented toward enhanced vectorization and to the use of object-oriented languages for the development of general-purpose multidimensional tracking subroutines

301

Response matrix methods revisited. Lewis, E E *Transactions of the American Nuclear Society (USA),* 60 339-340 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The physical basis for response matrix calculations is intuitively appealing The problem domain is divided into a number of coarse mesh nodes or cells The distributions of neutrons entering each node and of those exiting each node are approximated by a set of space angle basis functions defined along each interface Many variants of response matrices have been applied, with the level of approximation and the computational technique depending strongly on the physics of the particular class of problems under consideration Discrete ordinates, collision probabilities, Monte Carlo, and a number of other techniques have been utilized in the actual evaluation of response matrices Recently it has been demonstrated that coarse-mesh spherical harmonics approximations can be expressed as response matrices that obey nodal balance, and the need for iteration on the scattering source can be eliminated by formulating diamond-differenced discrete ordinates calculations as response matrices

302
Is there a strange attractor in your reactor?. Dorning, J *Transactions of the American Nuclear Society (USA),* 60 341 342 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco CA (USA) (26-30 Nov 1989)

The purpose of this paper is to provide an elementary introduction to certain concepts in bifurcation, nonlinear dynamics, chaos, and fractals that are relevant to - or are likely to become relevant to - the production of nuclear energy

303
Space-independent xenon oscillations revisited. Rizwan-uddin *Transactions of the American Nuclear Society (USA),* 60 343-345 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco CA (USA) (26-30 Nov 1989)

Recently, various branches of engineering and science have seen a rapid increase in the number of dynamical analyses undertaken This modern phenomenon often obscures the fact that such analyses were sometimes carried out even before the current trend began Moreover, these earlier analyses which even now seem very ingenuous were carried out at a time when the available information about dynamical systems was not as well disseminated as it is today One such analysis carried out in the early 1960s, showed the existence of stable limit cycles in a simple model for space-independent xenon dynamics in nuclear reactors The authors, apparently unaware of the now well-known bifurcation theorem by Hopf, could not numerically discover unstable limit cycles, though they did find regions in parameter space where the fixed points are stable for small perturbations but unstable for very large perturbations The analysis was carried out both analytically and numerically As a tribute to these early nonlinear dynamicists in the field of nuclear engineering, in this paper, the Hopf theorem and its conclusions are briefly described, and then the solution of the space-independent xenon oscillation problem is presented, which was obtained using the bifurcation analysis BIFDD code These solutions are presented along with a discussion of the earlier results

304
Optimal control of xenon concentration by observer design under reactor model uncertainty. Cho, Nam Z Yang, Chae Y , Woo, Hae S *Transactions of the American Nuclear Society (USA),* 60 369-370 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The state feedback in control theory enjoys many advantages, such as stabilization and improved transient response, which could be beneficially used for control of the xenon oscillation in a power reactor It is, however, not possible in nuclear reactors to measure the state variables, such as xenon and iodine concentrations For implementation of the optimal state feedback control law, it is thus necessary to estimate the unmeasurable state variables This paper uses the Luenberger observer to estimate the xenon and iodine concentrations to be used in a linear quadratic problem with state feedback To overcome the stiffness problem in reactor kinetics, a singular perturbation method is used

305
Configuration management - Design baseline is the key. Blocher, E *Transactions of the American Nuclear Society (USA),* 60 448 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The nuclear industry has concluded that because of the lack of understanding and use of design basis in day-to-day plant operations, erroneous decisions are made US Nuclear Regulatory Commission inspections, safety system functional inspections, and internal/external evaluations have identified that modifications and/or operating decisions have been made without sufficient engineering basis, which compromised safety system functionality A design-basis document (DBD) and comprehensive equipment list are recommended as the major elements of a design baseline program to support the organization with configuration data The system portion should create a data base and provide a cross reference for system functional requirements for the analyzed design-basis events (DBEs), minimal systems required to mitigate the DBE, and minimal components required for system functionality The DBD and equipment list must recognize the relationships between the information in the data base and the documents that support it As part of the change control requirements, a design baseline program must specify corporate interfaces, manage the data referenced, provide mechanisms to identify/classify open items and resolve open items

306
Properties of autoregressive model in reactor noise analysis, 2: Interpretation of poles in terms of power spectral density. Yamada, Sumasu, Kishida, Kuniharu, Sumita, Kenji *Journal of Nuclear Science and Technology (Tokyo) (Japan),* 27 No 8, 700-711 (Aug 1990)

Since the AR model widely used in the reactor noise analysis is mathematically a subset of the ARMA model, the characteristics of the AR model should be made clear for correct use of the resulting AR model A simple tool called the PSD contour-map associated with the normalized PSD chart is proposed for interpretation and evaluation of the poles of the AR model from the viewpoint of the PSD This tool not only visualizes the structure of the poles of the AR model but also provides us various information such as the importance of each pole of the AR model in the frequency region as well as the PSD difference of AR models with different model orders The fundamental properties of the PSD of a pair of complex conjugate poles are extracted from the PSD contour-map in the form of rules Model order reduction in AR model fitting is also discussed to show the effectiveness of the PSD contour-map It is also made clear in terms of the PSD that the information of the poles and zeros outside the convergence circle of the ARMA model is equivalently expressed in the AR model by relocation of the ring-poles representing the zero closest to the unit circle (author)

COMPONENTS AND ACCESSORIES

307
(CONF-9010185-9)
Improved eddy-current inspection for steam generator tubing. Dodd C V Pate J R (Oak Ridge National Lab TN (USA)) [1990] Contract AC05 84OR21400 9p NTIS PC A02/MF A01 OSTI INIS GPO Dep Order Number DE91002860
From 18 water reactor safety information meeting Gaithersburg MD (USA) (22 24 Oct 1990)
Oak Ridge National Laboratory has been engaged in the research and development of eddy current tests for a wide range of different problems Recent advances have been made on our multiple property techniques This technique generates a set of coefficients that correlate the readings from an eddy-current instrument to the properties of the test that produce the readings While this technique will work with reflection probes, pancake probes, or bobbin probes, we have concentrated on the latter since this type of test is the most widely used in the commercial inspection of steam generators The test properties varied include tube supports, tube sheets, copper deposits magnetite deposits, denting wastage pitting, cracking and IGA While our multiple property technique has given good results for several years, recent advances in personal computers have considerably improved the results Fits have been run for the differential bobbin probe that have included over 95 000 different sets of property values and their corresponding readings Multiple-property fits of these readings have given defect size fits with root-mean-square errors under 5% of the wall thickness for ASME Section XI standards Although the actual measurement of the defect depths is not that good (without corrections), the signal-to-noise ratio is very good even at copper and magnetite interfaces Different types of function fits have been tested for the various types of probes and defects, and optimum functions have been determined for each 3 refs , 8 figs , 1 tab

308
(EPRI-NP-6377-M)
Guidelines for the design and operation of makeup water treatment systems. Lee, Y H , Planek, M A , Sopocy, D M , Tomaga, C M , Abrams, I M , Anderson, C C , Balazs, M K , Houskava, J , Williams, R (Electric Power Research Inst , Palo Alto, CA (USA), Sargent and Lundy, Chicago, IL (USA), Balazs Analytical Lab , Mountain View, CA (USA)) Jun 1989 23p Research Reports Center Box 50490 Palo Alto CA 94303
These guidelines present the industry with a standardized program to ensure the optimum design and operation of their individual makeup water treatment systems These guidelines present in a non technical and non proprietary format the makeup water treatment system design and operating topics that are discussed in detail in Volumes 1 and 2 of NP 6377 SL The individual guidelines contained in Volumes 1 and 2 are presented as separate imperative statements followed by a technical justification discussion which provides further explanations In addition and when applicable the guidelines relate pertinent operational in regard to monitoring parameters for operation, alternative actions, troubleshooting, management responsibilities and shutdown practices Design considerations are also addressed, when applicable, in regard to equipment cost and advantages and disadvantages for the design recommendations Appendices provide background information for performance criteria, component description, economic evaluation procedures and definitions 4 refs

309
(EPRI-NP-6997-M)
Alloy 690 for steam generator tubing applications. Gold R E , Harrod, D L , Aspden R G , Baum A J (Electric Power Research Inst , Palo Alto, CA (USA), Westinghouse Electric Corp , Pittsburgh PA (USA)) Oct 1990 55p Research Reports Center, Box 50490, Palo Alto, CA 94303
This report has been prepared to provide background information for Ni-Cr-Fe Alloy 690 which is currently the material of choice for steam generator heat transfer tubing applications Activities directed toward the qualification of Alloy 690 for these applications are summarized, this includes efforts which focused on optimization of materials procurement specifications Emphasis is placed on research accomplished primarily in the four year period from June 1985, the time of the first EPRI Workshop on Alloy 690 was held The topic is treated in a broad sense, and includes review of the physical metallurgy of the alloy, tube manufacturing processes, the properties of commercial production tubing, and the corrosion behavior of Alloy 690 in environments appropriate to steam generator service 12 refs , 7 figs , 8 tabs

310
(EUR-12389)
***Structural reliability benchmark* exercise for primary-circuit components life prediction methods.** Lehrke H P (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 17p NTIS (US Sales Only) PC A03/MF A01
Results of NDT inspections, crack growth data taken from small specimens, and results of stress calculation together with a load time history are used to predict the growth to failure of defects intentionally introduced into the welds of a 1/5 scale pressure vessel The prediction results are compared with further NDT results taken during the fatigue test Analysing the discrepancies between the analytical and the test results, the question, how to define an equivalent crack to each detected defect, is found to be the weakest link in the procedure of crack growth and life prediction The NDT results differ considerably from one inspection to another and contain only the positions and sizes of the defects but no information on the shape and whether a defect is crack-like or not Therefore it is necessary to define equivalent cracks with assumed shapes e g elliptical, and to take into account an incubation time of crack initiation at the blunt (instead of sharp) edges of the defects However, NDT gives no information to set up the necessary calculation parameters, especially for the decision, whether a defect of critical size will start to grow immediately when in-service, after a short (or long) incubation time, or never

311
(EUR-12390)
Analysis of failures in concrete containments. Moreno-Gonzalez, A (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 160p NTIS (US Sales Only), PC A08/MF A01

The function of Containment, in an accident event, is to avoid the release of radioactive substances into the surroundings Containment failure, therefore, is defined as the appearance of leak paths to the external environment These leak paths may appear either as a result of loss of leaktightness due to degradation of design conditions or structural failure with containment material break This document is a survey of the state of the art of Containment Failure Analysis It gives a detailed description of all failure mechanisms, indicating all the possible failure modes and their causes, right from failure resulting from degradation of the materials to structural failure and linear breake failure Following the description of failure modes, possible failure criteria are identified, with special emphasis on structural failure criteria These criteria have been obtained not only from existing codes but also from the latest experimental results A chapter has been dedicated exclusively to failure criteria in conventional structures, for the purpose of evaluating the possibility of application to the case of containment As the structural behaviour of the containment building is very complex it is not possible to define failure through a single parameter It is therefore advisable to define a methodology for containment failure analysis which could be applied to a particular containment This methodology should include prevailing load and material conditions together with the behaviour of complex conditions such as the liner-anchorage-cracked concrete interaction

312
(EUR-12391)
Modelling of cracking and inelastic behaviour of reinforced concrete structures. Young A G , Albana, M O (Commission of the European Communities, Luxembourg (Luxembourg)) Sep 1989 47p NTIS (US Sales Only), PC A03/MF A01

The report contains a review of work available in the literature on local bond transfer and the main factors which influence it, involving deformed reinfocing bar Possible load transfer mechanisms are investigated and the significance of secondary cracking, local consolidation and shearing assessed On the basis of these studies a linkage element which realistically models bond action, and is applicable to both monotonic and cyclic load, is proposed Its ability to accurately predict stress, strain and crack geometry in typical reinforced concrete components is demonstrated by comparison of the results of finite element analysis using this model with experimental data Aspects requiring further research are identified An analysis of the dynamic response of a reinforced concrete beam is given which makes the simplifying assumption of rigid-plastic behaviour A comparison of the analytical solution with experimental results obtained by bend tests in the Large Dynamic Test Facility at Ispra shows that, despite the neglect of elastic vibrations, a reasonable prediction of the fundamental response is obtained providing due allowance is made for rate-of-strain effects

313
(EUR-12392)
Development of realistic concrete models including scaling effects. Carpinteri, A (Commission of the European Communities Luxembourg (Luxembourg)) Sep 1989 67p NTIS (US Sales Only), PC A04/MF A01

Progressive cracking in structural elements of concrete is considered Two simple models are applied, which, even though different, lead to similar predictions for the fracture behaviour Both Virtual Crack Propagation Model and Cohesive Limit Analysis (Section 2), show a trend towards brittle behaviour and catastrophical events for large structural sizes A numerical Cohesive Crack Model is proposed (Section 3) to describe strain softening and strain localization in concrete Such a model is able to predict the size effects of fracture mechanics accurately Whereas for Mode I, only untieing of the finite element nodes is applied to simulate crack growth, for Mixed Mode a topological variation is required at each step (Section 4) In the case of the four point shear specimen, the load vs deflection diagrams reveal snap-back instability for large sizes By increasing the specimen sizes, such instability tends to reproduce the classical LEFM instability Remarkable size effects are theoretically predicted and experimentally confirmed also for reinforced concrete (Section 5) The brittleness of the flexural members increases by increasing size and/or decreasing steel content On the basis of these results, the empirical code rules regarding the minimum amount of reinforcement could be considerably revised

314
(EUR-12394)
Review of constructive models for concrete. Xiaoping, Y , Ottosen, N S , Thelandersson, S , Nielsen, M P (Commission of the European Communities, Luxembourg (Luxembourg)) Nov 1989 328p NTIS (US Sales Only), PC A15/MF A01

This report has been prepared for the Commission of the European Communities, Joint Research Centre, ISPRA The report reviews the constitutive models for concrete and is a part of a survey of the status of the analytical capabilities for predicting the structural response of NPP concrete containment buildings to severe loading conditions

315
(EUR-12395)
Analytical capability for predicting structural response of NPP concrete containments to severe loads. Planas, J , Guinea, G , Trbojevic, V M , Marti, J , Martinez, F , Cortes, P (Commission of the European Communities, Luxembourg (Luxembourg)) Dec 1989 243p NTIS (US Sales Only), PC A11/MF A01

A survey has been conducted on the state-of-the-art of analytical techniques for predicting the structural response of concrete containment buildings under severe accident conditions The validity of inelastic analysis is often limited by the inadequacy of the material models adopted This is specially true in the case of materials which undergo localization phenomena in the course of the deformation process Because of this the Joint Research Centre at Ispra has given a high priority to the review of existing constitutive models for concrete Such models must be able to describe concrete behaviour with and without steel reinforcement across the complete stress range, from initial elastic behaviour to and beyond the point of failure For reinforced and prestressed concrete, segregated models (where concrete and steel are independently simulated) are preferred A review of existing constitutive models for mass concrete has been conducted The review focused on necessary features for describing the near-peak and post-peak stages of deformation Special attention was dedicated to the localization of strains in tension and the post-peak softening behaviour Existing models for representing the concrete steel bond were also reviewed These models are still relatively simplistic and incorporate seldom a number of effects of considerable importance sustained, dynamic and cyclic loading, environmental effects, etc Finally, the computational procedures currently available for modelling problems involving the ultimate capacity of concrete containments have also been reviewed This includes methodologies for modelling amongst other mass concrete, cracking procedures, bond behaviour, in existing computer codes

316
(EUR–12518)
Pre-stressed concrete reactor vessel with built-in planes of weakness. Dawson, P , Paton, A A , Fleischer, C C (Commission of the European Communities, Luxembourg (Luxembourg)) 1989 62p NTIS (US Sales Only), PC A04/MF A01

This report describes a study that has been carried out to extend previous work to investigate the feasibility of constructing regions of pre-stressed concrete reactor vessels (PCRV) and biological shields which become activated using easily removable blocks, separated by a suitable membrane The previous study concluded that, from preliminary analyses, such a concept appeared feasible and recommended that further work should be done The present study was therefore commissioned to carry out more detailed analytical work and to complement this with the design, construction and pressure testing of two small-scale, single-cavity PCRV models, one without planes of weakness and one with planes of weakness immediately behind the cavity liner The report describes the analyses, the model design, construction and testing, and presents relevant results It concludes that the planes of weakness concept could offer a means of facilitating the dismantling of activated regions of PCRV, biological shields and similar types of structure

317
(EUR–12578, pp 585-591)
Equipment Maintenance management support system based on statistical analysis of maintenance history data. Shimizu, S , Ando, Y , Morioka, T (Nippon Atomic Industry Group Co , Ltd Tokyo (JP)) 1990 695p NTIS (US Sales Only), PC A99/MF A01 Order Number DE91719164 (CONF-8904156–, CSNI–159)

From CSNI-CEC specialist meeting on trends and pattern analyses of operational data from nuclear power plants, Rome (Italy) (3-7 Apr 1989)

In Trend and pattern analyses of operational data from nuclear power plants

Plant maintenance is recently becoming important with the increase in the number of nuclear power stations and in plant operating time Various kinds of requirements for plant maintenance, such as countermeasures for equipment degradation and saving maintenance costs while keeping up plant reliability and productivity, are proposed For this purpose, plant maintenance programs should be improved based on equipment reliability estimated by field data In order to meet these requirements, it is planned to develop an equipment maintenance management support system for nuclear power plants based on statistical analysis of equipment maintenance history data The large difference between this proposed new method and current similar methods is to evaluate not only failure data but maintenance data, which includes normal termination data and some degree of degradation or functional disorder data for equipment and parts So, it is possible to utilize these field data for improving maintenance schedules and to evaluate actual equipment and parts reliability under the current maintenance schedule In the present paper, the authors show the objectives of this system, an outline of this system and its functions, and the basic technique for collecting and managing of maintenance history data on statistical analysis It is shown, from the results of feasibility tests using simulation data of maintenance history, that this system has the ability to provide useful information for maintenance and the design enhancement

318
(IEN-DITRA–003/89)
Sodium as a reactor coolant. Cesar, S B G (Instituto de Engenharia Nuclear (IEN), Rio de Janeiro, RJ (Brazil)) 6 Apr 1989 13p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607636

This work is related to the use of sodium as a reactor coolant, to the advantages and problems related to its use, its mechanical, thermophysics, eletronical, magnetic and nuclear properties It is mainly a bibliographic review, with the aim of gathering the necessary information to persons initiating in the study of sodium and also as reference source (author)

319
(KI–1989-3)
Methods for combating microorganisms in cooling water systems - a literature study and a market inventory. Thierry, D (Swedish Corrosion Inst , Stockholm (Sweden)) 20 Jul 1989 23p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607637

One of the greatest current problems in both closed and open cooling water systems is that of micro- and macro-organisms In view of the environmental effects associated with the discharge of chemicals, the range of biocides and alternative methods for combating micro-organisms has increased during recent years This report presents a brief description of the organisms which contribute to corrosion problems and the mechanisms associated with microbial corrosion Thereafter descriptions are given of 15 different biocides which are used in both open and closed cooling systems In each case, details are given of their chemical compositions and mode of action and of their effects on metals and on the environment Finally, alternative methods of combating micro-organisms in cooling water systems are briefly described The report also includes a survey of the biocides for cooling water systems which are available on the Swedish market (author)

320
(NUREG/CP–0111)
Proceedings of the symposium on inservice testing of pumps and valves. (Nuclear Regulatory Commission, Washington, DC (USA) Office of Nuclear Reactor Regulation, EG and G Idaho, Inc , Idaho Falls, ID (USA)) Oct 1990 Contract AC07-76ID01570 469p (EGG–2609, CONF-890852–) NTIS, PC A20/MF A01 GPO, OSTI, INIS

From ASME/NRC symposium on inservice testing of pumps and valves, Washington, DC (USA) (1-3 Aug 1989)

The 1990 Symposium on Inservice Testing of Pumps and Valves, jointly sponsored by the Board on Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the Nuclear Regulatory Commission, provided a forum for the discussion of current programs and methods for inservice testing at nuclear power plants The symposium also provided an opportunity to discuss the need to improve inservice testing in order to ensure the reliable performance of pumps and valves The participation of industry representatives, regulators, and consultants resulted in the discussion of a broad spectrum of ideas and perspectives regarding the improvement of inservice testing of pumps and valves at nuclear power plants

321
(SAND-90 1690C)
Mechanical properties of cables exposed to simultaneous thermal and radiation aging. Jacobus M J, Fuehrer, G F (Sandia National Labs, Albuquerque NM (USA)) [1990] Contract AC04-76DP00789 16p (CONF-9010185-5) NTIS, PC A03/MF A01 - OSTI, GPO Dep Order Number DE91001769

From 18 water reactor safety information meeting, Gaithersburg, MD (USA) (22-24 Oct 1990)

Sandia National Laboratories is conducting long-term aging research on representative samples of nuclear power plant Class 1E cables The objectives of this program are to determine the suitability of these cables for extended life (beyond the 40 year design basis) and to assess various cable condition monitoring (CM) techniques for predicting remaining cable life This paper provides the results of mechanical measurements that were performed on cable specimens cross linked polyethylene neoprene jackets chlorinated polyethylene jackets fiberglass braid jackets and chlorosulfonated polyethylene jackets aged at relatively mild simultaneous thermal and radiation exposure conditions for periods of up to nine months After aging, some of the aged samples, as well as some unaged samples, were exposed to accident gamma radiation at ambient temperature The mechanical measurements discussed in this paper include tensile strength, ultimate elongation, and compressive modulus 10 refs, 22 figs, 2 tabs

322
Integrated flow and heat removal performance evaluation for cooling water systems in nuclear power generating stations. Shih, M, Larssen, K O, Mody, J, Hoffmann, M W pp 1063-1072 of Power-gen 1989 Conference papers, Volumes V and VI Houston, TX (US), Power-Gen (1989) 413p (CONF-891217-)

From POWER-GEN '89 2nd conference and exhibition for the power generation industries, New Orleans, LA (USA) (5-7 Dec 1989)

A cooling water system in a power plant installation will experience a number of changes in effectiveness and performance during its lifetime Some of the changes will occur gradually, such as through aging, or fouling, of the internal surfaces Other changes will occur suddenly, such as through component failures and replacement, system modifications and maintenance Throughout these changes the plant owner must assure himself that minimum design requirements are met in order to achieve personnel safety and a sustained design generating output In the case of a nuclear power plant, the ability of a safety-related system, such as the component cooling water (CCW) system, to meet its minimum design requirements must be closely monitored and maintained to keep the plants ability to safely shut down, following a design basis accident (DBA) from being compromised It is believed that an integrated system flow and heat removal performance evaluation provides vital data necessary to preserve system design bases by focusing appropriate maintenance activities where needed most This paper describes the integrated flow and heat removal analysis approach and the IBM PC based programs used to greatly simplify the task of establishing the CCW system performance for any given set of system parameters The program was used to establish a consistent set of system design data and to consolidate a number of previously developed manual design calculations and studies for the Trojan Nuclear Plant It was also and will continue to be used to study the effects of various system perturbations on overall performance

323
DATYS integrates piping and supports engineering. Rendon, J G, Fraile, A R *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 57-58 (Jul 1990)

Empresarios Agrupados of Spain has developed an interactive software package which computerizes and integrates the whole range of tasks involved in pipework engineering, including drawing, design, analysis and support calculations Its strength lies in its modularity and in the ability to re-evaluate and modify existing projects (author)

324
Improving safety margins in pressure transmitters. *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 26-27 (Jul 1990)

A 1989 failure report highlighted the need for reliable sealing in electronic pressure transmitters used in nuclear plants - a defective seal can cause leakages from the sensor capsule Statham Transducer has taken this on board and modified its transmitters to improve their sensor seal integrity (author)

325
PRA and the common-mode problem. Ericson, D M *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 431, 38-40 (Jun 1990)

There is solid recognition that common cause/common mode failure (CCF) plays a significant role in nuclear power plant risk and risk assessment But there is also agreement that the quality of the data used to develop CCF analyses can be improved upon (author)

326
Adapting ultrasonics to examine PWSCC in steam generators. Dobbeni, D, Degreve, D *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 430 46 48 (May 1990)

Field and laboratory results carried out by Laborelec have confirmed the benefits of applying ultrasonic techniques for detecting and sizing small volume cracks like Primary Water Stress Corrosion Cracking (PWSCC) In particular the small focal point of the ultrasonic beam has a clear advantage when detecting circumferential PWSCC in the presence of multiple axial cracks (author)

327
Innovative projects of SIGMA Corporation, Olomouc. Koecher, J *Strojnicky Casopis (Czechoslovakia),* 41 No 1, 3-8 (1990) (In Czech)

Plans are outlined of future production of SIGMA Corp, Czechoslovakia For power generation, pumps and fittings and also high-pressure pipes for thermal power plants at above-critical parameters will be produced, production of equipment for nuclear power plants will include new types of pumps and special fittings for domestic as well as foreign light water reactors and fast reactors, with special regard to reliability and lifetime (P A) 5 figs

328
Self-starting of auxiliary electric motors at NPPs and the ways for its improvement. Chernovets, A K, Semenov, K N, Shargin, Yu M, Mel'nichnikov, S A, Chizhkov, K G *Ehlektricheskie Stantsii (USSR),* No 10, 21-29 (Oct 1989) (In Russian)

Modes of self-starting of auxiliary electric motors at NPPs with the WWER-1000 and RBMK-1000(1500) type reactors are considered It is shown that transition at the first stage of transformation to the voltage of 10 kV using the transformer with capacity of 63MVxA at small voltage of short circuit U_{sc} is the solution of the self-starting problem At the second stage of transformation the problem of the self starting is successfully solved by coordination of the second stage transformer capacities their loading during self-starting and the value of U_{sc}

329
On the structure of moisture-steam flow in by-pass receivers of an intermediate separation and steam superheating system. Filippov, G A, Nazarov, O I, Filimonova, V P, Antoshkin, N A, Robinov, A V *Ehnergomashinostroenie (USSR),* No 6, 24-27 (Jun 1989) (In Russian)

Results of experimental investigations into the processes in receivers of systems of intermediate steam separation and superheating in turbine plants built at Kola Novovoronezh and Zaporozhe NPPs are presented Based on the obtained data analysis a notable stratification of a two-phase flow and moisture concentration in the near-wall region are revieled The flow stratification is especially sufficient in receivers with smooth bends and can achieve 90% If straight bends with guiding blades are used in the receivers the flow stratification would be lower and the flow nucleus would contain about 50% of the designed moisture quantity

330
Film separators in receivers of NPP turbine plants. Sorokin, Yu L, Demidova, L N, Sidorov, V N *Ehnergomashinostroenie (USSR),* No 6, 28-31 (Jun 1989) (In Russian)

Principle design layouts of film separators used in the receivers in front of steam separators-superheaters of NPP saturated-steam turbine plants are considered Data characterizing the film separator efficiency dependence on their structural features are presented

331
Efficiency of a separator in the system of intermediate separation and steam superheating. Khrunich, A N, Gostev, D G *Ehnergomashinostroenie (USSR),* No 6, 31-34 (Jun 1989) (In Russian)

Structural featires and results of testing the film separator with in-vessel steam circulation are considered It is proved that application of film separator allows one to increase the efficiency of intermediate separation and steam superheating system The efficiency of the film separator itself can achieve 30-90% Depending on the place of the film separator mounting in the system and configuration of a turning bend in front of it the turbine plant electric capacity can be increased by 1 87-4 MW

332
Operational conditions for heat-transfering surfaces in steam separators-superheaters for condenser vertical arrangement. Mukhachev, V L, Panin, V V *Ehnergomashinostroenie (USSR),* No 6, 34-37 (Jun 1989) (In Russian)

Operating conditions for heat-transfering surfaces in separators-superheaters with modular and cassete designs are investigated It is shown that the presence of hydraulic and heat variations over the cross section of separators-superheaters creates unstable operating conditions It is proved that throttling and drainage of steam superheaters by heating steam allow one to reduce the effect of heat and hydraulic variations to minimum

333
Optimization for pipe support locations. Kishida, Kazuo, Yamadera, Masao *Ishikawajima Harima Giho (Ishikawajima-Harima Engineering Review) (Japan),* 30 No 4 230 233 (Jul 1990) (In Japanese)

Currently, pipe support locations are determined by engineering judgement and professional experience of pipe design engineers, which are then verified by piping stress analyses This process requires a lot of man-hours and more experienced engineers to fix satisfactory locations and types of supports for many meters of pipes in power generation plants For resolution of the problem, a pipe support optimization computer program was introduced The program can determine the optimal support configurations and types automatically, using the iteration process that a single support is added one by one for a free support model of piping system, in cases where this effect is evaluated It was built in the existing pipe design CAD system to be more helpful to the engineers The system enabled the engineers to do more reliable pipe support design (author)

334
Mechanisms of fracture in ferritic steels and implications for structural applications in nuclear systems. Lucas, G E *Transactions of the American Nuclear Society (USA),* 60 291-292 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Ferritic steels are widely used in pressure vessels for light water reactors and have potential applications in both advanced fission and fusion reactors While they have advantages particular to each application, they exhibit a transition in fracture mode from ductile to cleavage fracture with decreasing temperature Degradation of ductile fracture resistance or increases in this transition temperature with neutron irradiation can limit the lifetime of the structure Understanding the mechanisms of fracture can improve both the ability to forecast these changes and the way in which fracture behavior is incorporated in design Hence, it is the purpose of this paper to review the mechanisms of both ductile and cleavage fracture in ferritic steels For design purposes, the conditions under which preexisting defects or cracks will propagate in a structure under load must be known and avoided For component design linear elastic and even elastic-plastic fracture mechanics may provide inaccurate estimates of the load-bearing capability of a defected structure Instead, finite element analyses of the stress and strain fields around the defect combined with appropriate mechanistic descriptions of the conditions required for fracture initiation would provide a better approach

335
Sequential tests for failure detection. Pazsit, I *Annals of Nuclear Energy (UK),* 17 No 7, 347-352 (1990)

The problem of constructing sequential tests for fault detection methods that are based on the concept of Mahalanobis distance is addressed First the distribution of the Mahalanobis distance that belongs to the degraded (alternative) state of a multivariate system is determined Then three sequential tests are constructed and their performance investigated for fault detection For each test, the average sample number and an upper limit for the number of samples necessary so that the test will terminate with a given probability are calculated (author)

336
A new approach to structural integrity assessment for ductile fracture. Dong, Y M , Yang, W , Hwang, K C *Fatigue and Fracture of Engineering Materials and Structures (UK),* 13 No 4, 399-410 (1990)

A new approach to structural integrity assessment based on ductile fracture is explored in the present paper A simplified, yet more convenient methodology than the conventional EPRI elastic-plastic fracture analysis is outlined which assesses the ductile fracture instability of structures by an intersection rather than a tangent construction The J-equivalence principle of crack growth is employed for tearing instability determination of single edge cracked plates An admissible stress curve method is established with a peak value indicating instability This approach gives reasonable predictions concerning the burst pressure of preflawed pressure vessel A theoretical J resistance curve, which is justified by experimental measurements for steels with relatively high yield strength, can be incorporated for a complete analytical characterization of the defect assessment procedure (author)

337
Plant valve alignment changes: Should they be included in plant configuration management processes?. Lentine, F G , Brecken, D R *Transactions of the American Nuclear Society (USA),* 60 450-452 (1989) (CONF 891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit San Francisco CA (USA) (26 30 Nov 1989)

Configuration management is the process that a utility uses to answer the following questions (1) what is the current physical plant configuration, (2) what should be the current plant configuration, (3) is the utility able to prove their configuration (by readily providing supporting documentation), (4) does the utility have a process in place to maintain their configuration management program? Many utilities are developing configuration management programs Often, these programs are being developed, engineered, and managed by utility design engineering departments and/or their architect/engineers In most cases, these programs address long-term physical plant modifications and do not account for plant operational changes Few utilities have perceived the potential pitfalls of this deficiency and the ramifications to plant safety/operations if an original system operational alignment is altered, based only on good operational experience or perceived benefit This paper presents three cases, using Zion nuclear generating station, that illustrate the need to incorporate the operational aspects

338
Application of electronic market technology to the electric utility industry. Kelly, J C *Transactions of the American Nuclear Society (USA),* 60 463-465 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

This paper presents a report on progress made in 1989 toward applying electronic market technology to the electric utility industry that enables widespread display and retrieval of inventory availability data The computerized electronic market system is described and an experience report is presented on the use of a similar system by airlines, aircraft equipment suppliers, and aviation service firms to pool replacement component availability data for their mutual benefit and the support of an efficient market in replacement components The application of large scale electronic market technology to the electric utility industry is a significant new development

339
Technical considerations in snubber reduction programs. Longo, D L , Kitz, G T *Transactions of the American Nuclear Society (USA),* 60 468 469 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Snubber reduction has been a much discussed topic for the past several years The advantages to the owner in terms of reduced radiation exposure, lower operating costs, and better plant maintainability are obvious but difficult to quantify The costs associated with performing a snubber reduction program can be quantified, but can vary widely depending on how the program is implemented One approach is to divide the scope of the work into several parts, based on similarities and differences of the plant systems and technical requirements for example, for this program, the work was divided into three parts nuclear steam supply system, torus-attached piping, and balance of plant This approach facilitates addressing the variations in design requirements and the variations in snubber removal priorities that are related to each plant segment It also allows the experience gained in the first part to be applied to the subsequent parts This paper describes one utility's experience in successfully completing two parts of a three-part snubber reduction program and focuses on the various technical issues that must be addressed along with those areas where the utility should establish checkpoints that are critical to the success of the overall program It also describes how the experience from the first two parts will be factored into the final part of the snubber reduction program

340
Component monitoring during operation in nuclear power stations. Sacher, H *VGB Kraftwerkstechnik (Germany, F R),* 70 No 9, 773-777 (Sep 1990) (In German)

In the design of nuclear power stations all stresses on the pressure-bearing enclosure in respect of their effect on component strength and integrity are investigated In addition, local and global monitoring during operation is carried out, in order to detect also those stresses which have been hitherto unknown or are caused by malfunctions of components In this way a realistic assessment of the condition of the pressure-bearing enclosure is possible and any irregularities which can imply malfunctions of components or physical effects not hitherto known can be detected early (orig)

341
Fire extinguishing device for nuclear power plant. Arakawa Ken (to Toshiba Corp Kawasaki Kanagawa (Japan) Nippon Atomic Industry Group Co Ltd Tokyo (Japan)) Japan Patent 2 136790/A/ 25 May 1990 Filed date 17 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

Fire extinguishing pipelines disposed in turbine buildings of low earthquake proof grade and fire extinguishing pipelines disposed in reactor buildings of high earthquakes proof grade have been used in common with each other Accordingly if the fire extinguishing device in the turbine buildings designed for low earthquake proof grade are partially destroyed upon occurrence of medium-scale earthquakes, there is a worry that fire extinguishing water can not be supplied to the inside of the reactor buildings In view of the above an emergency fire extinguishing water system using a fire extinguishing reservoir at the outdoor of low earthquake proof grade as a feedwater source and suitable to the low earthquake proof grade is disposed in the turbine buildings Another emergency fire extinguishing water system using an emergency fire extinguishing water reservoir disposed in the reactor buildings as a feedwater source and suitable to the high earthquake proof grade is disposed in the reactor buildings Then, ordinary fire extinguishing water system and the emergency fire extinguishing water system are connected to each other Thus upon occurrence of earthquakes if the function of the ordinary fire extinguishing water system of low earthquake proof grade is lost fires breaking out in the reactor buildings can rapidly be extinguished (N H)

342
Plant abnormality inspection device. Takenaka, Toshio (to Mitsubishi Electric Corp , Tokyo (Japan)) Japan Patent 2-136793/A/ 25 May 1990 Filed date 16 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

The present invention concerns a plant abnormality inspection device for conducting remote or automatic patrolling inspection in a plant and, more particularly, relates to such a device as capable of detecting abnormal odors That is the device comprises a moving device for moving to a predetermined position in the plant a plurality of gas sensors for different kind of gases to be inspected mounted thereon, a comparator for comparing the concentration of a gas detected by the gas sensor with the normal gas concentration at the predetermined position and a judging means for judging the absence or presence of abnormality depending on the combination of the result of the comparison and deliverying a signal if the state is abnormal As a result a slight amount of gas responsible to odors released upon abnormality of the plant can be detected by a plurality of gas sensors for different kinds gases to rapidly and easily find abnormal portions in the plant (I S)

343
Leak processing system for valve gland portion. Nishino Masami (to Toshiba Corp , Kawasaki, Kanagawa (Japan), Toshiba Engineering Co Ltd , Tokyo (Japan)) Japan Patent 2-136795/A/ 25 May 1990 Filed date 18 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

When a process fluid for a valve to be checked is at such a normal temperature as during reactor operation, leaked fluid can be detected depending on the temperature increase accompanying the leakage However, detection is difficult if the temperature of the process fluid for the valve to be checked is low and, if leakage is detected after the reactor start-up, repair has to be applied after the shutdown of the plant Then gland leak is detected by detecting the pressure instead of the temperature in the pipeline system and the leak flow rate is calculated based on the pressure As a result leakage is detected irrespective of the temperature of the leaked fluid and for instance, leakage can be detected even in a case where the temperature is not high as in the case of pressure proof test for the pressure vessel before start-up It can contribute much to the improvement of the plant operation efficiency and can determine the leak flow rate at a high accuracy (N H)

344
Fuel inspection device. Tsuji, Tadashi (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-134595/A/ 23 May 1990 Filed date 16 Nov 1988 3p (In Japanese) JAPIO Also available from INPADOC

The fuel inspection device of the present invention has a feature of obtaining an optimum illumination upon fuel rod interval inspection operation in a fuel pool That is an illumination main body used underwater is connected to a cable which is led out on a floor A light control device is attached to the other end of the cable and an electric power cable is connected to the light control device A light source (for example incandescent lamp) is incorporated in the casing of the illumination main body and a diffusion plate is disposed at the front to provide a plane light source The light control device has a light control knob capable of remote-controlling the brightness of the light of the illumination main body In the fuel inspection device thus constituted halation is scarcely caused on the image screen upon inspection of fuels by a submerged type television camera to facilitate control upon inspection Accordingly, efficiency of the fuel inspection can be improved to shorten the operation time (I S)

345
Ultrasonic imaging device. Takahashi, Akio (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-134594/A/ 23 May 1990 Filed date 16 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

The present invention concerns a ultrasonic imaging device used for confirming the separation of control rods upon fuel exchange or detecting the rise of reactor core constituent elements in a FBR type reactor, which can reduce thermal shocks to a transducer and improve the reliability That is, the shape of the portion for mounting the transducer is made spherical with the transducer being as a center With such a constitution, since radiant heat from the a reactor core is reflected at the mounting portion and concentrated to the transducer, preheating time in a cover gas can be shortened Accordingly, the thermal shocks to the transducer can be moderated to provide a ultrasonic imaging device at a high reliability Further, the fuel exchange time can be shortened by decreasing the preheating time for the transducer, which can contribute to the improvement of the reactor operationability (I S)

346
Fuel storage rack. Hanawa, Keiji (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-129598/A/ 17 May 1990 Filed date 10 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

If fuel enrichment degree and reactivity are increased remarkably with an aim of effective utilization of fuels in a nuclear power plant, it may become difficult to ensure subcriticality for a spent fuel storage rack now in use It may be considered to replace the already disposed fuel storage rack with other rack of sufficient subcriticality, however, there are disadvantages that operators suffer from radiation exposure during handling of a fuel pool or fuel storage rack contaminated with radioactivity and the replacement operation needs a considerable cost In view of the above, neutron absorbing plates are disposed detachably on the inner surface of the main body of the fuel storage rack They can be disposed by using cranes, etc from the outside of the fuel pool while storing spent fuels as they are Accordingly it is not only economical in view of continuous usage and easy operation but also the operator s expo sure dose can be reduced (N H)

347
Reactor building. Ebata Sakae (to Hitachi Ltd , Tokyo (Japan)) Japan Patent 2-128199/A/ 16 May 1990 Filed date 8 Nov 1988 3p (In Japanese) JAPIO Also available from INPADOC

At least one valve rack is disposed in a reactor building, on which pipeways to a main closure valve, valves and bypasses of turbines are placed and contained The valve rack is fixed to the main body of the building or to a base mat Since the reactor building is designed as class A earthquake-proofness and for maintaining the S_1 function, the valve rack can be fixed to the building main body or to the base mat With such a constitution, the portions for maintaining the S_1 function are concentrated to the reactor building As a result the dispersion of structures of earthquake proof portion corresponding to the reference earthquake vibration S_1 can be prevented Accordingly the conditions for the earthquake proof design of the turbine building and the turbine/electric generator supporting rack are defined as only the class B earthquake-proof design conditions In view of the above, the amount of building materials can be saved and the time for construction can be shortened (I S)

348
Atmosphere cooling device in reactor. Sonoda, Takayuki, Tada, Kenji, Miyagawa, Takenao (to Hitachi Ltd , Tokyo (Japan)) Japan Patent 2-126190/A/ 15 May 1990 Filed date 2 Nov 1988 4p (In Japanese). JAPIO Also available from INPADOC

In an atmosphere cooling device for cooling air in a reactor container and circulating it by a blower, since warm air has to be sent over a long distance from the upper portion of the reactor container to a ventilation port, the temperature of the warm air is lowered during sending and the temperature control for the inside of a pedestal is extremely difficult Then, according to the present invention, means for supplying not cooled air is disposed inside of the pedestal below the pressure vessel Air cooled by a cooler is supplied to the upper portion of the reactor container The not cooled air is supplied to the pedestal below the reactor pres sure vessel As a result the upper portion of the pressure vessel with a great amount of heat generation is cooled by the cooled air, which does not cause a low temperature portion in the pedestal to prevent the elevation of the relative humidity Accordingly, the temperature in the reactor container can be unified and the relative humidity in the pedestal can be suppressed to maintain a favorable circumstance for stainless steels (N H)

349
Leak monitoring device for high temperature-high pressure vessel. Kato, Shinya, Tsuneoka, Osamu, Shiga, Shigenori, Okazaki, Hideyuki (to Nippon Atomic Industry Group Co Ltd , Tokyo (Japan), Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-126195/A/ 15 May 1990 Filed date 4 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

The purpose of the present invention is to always monitor the leakage from a pressure vessel used under high temperature and high pressure circumstance during plant operation In view of the above, the device of the present invention comprises a pipeway for sucking gases in the circumstance where the leakage occurs and an alarm device for judging the occurrence of the leakage by signals from a sensing device for monitoring the ingredients of the sucked gases With this constitution, since a leak monitoring device is disposed around a pressure vessel at high temperature and high pressure where the leakage may possibly occur, leakage can be detected at an early stage and the amount of the leakage can be evaluated quantitatively For the sensor used in the device, a sensor suitable for the materials present in the pressure vessel is used in addition to a temperature sensor, liquid level sensor and radiation sensor For example, if nuclides are analyzed by a radiation sensor as a sensing method inherent to a nuclear power plant, leakage of water from the reactor core can be detected (I S)

350
An integrated heat-distributing system of nuclear and conventional power plants. Krizek, V , Kral, B Czechoslovakia Patent 264401/B1/ 15 Nov 1989 4p (In Czech) INIS

An integral heat distribution system is designed for nuclear as well as fossil-fuel power plants It comprises at least one takeoff piping leading from the turbine body and ending in the steam heating space This space contains a heat exchange body for the heat distribution which is integrated with the recovery heat-exchange body It is recommended that the heating condensate takeoff system be constituted by a separate collector with piping for the recovery section and a separate collector with piping for the heat distribution section This heat distribution system simplifies the heating and piping design of the turbine hall and in some instances enables additional heat takeoff, provided that adaptations of the turbine hall are made (P A) 1 fig

351
A tubular beam spacing grid. Plihal, K , Matal, O , Urbanek, M Czechoslovakia Patent 263640/B1/ 14 Aug 1989 9p (In Czech) INIS

The spacing grid consists of form strips with a central planar segment fitted with technological holes and lying always between two rows of tubes in a plane parallel to the longitudinal axis of the tubes This segment is followed by fastening elements and a pair of spacing elements The spacing between the spacing elements is identical with that between the tubes in a tube row of the bundle At least on one side, the central planar segment is reinforced with a perforated strip The form strips with the planar segment and the pair of spacing elements can be fabricated in mere two operations, which enables the spacing grids to be manufactured with a high precision The tubes in the spacing grid region can be washed along their perimeter The strength of the grid can be increased by inserting supporting pins The design may find particular use in nuclear power plant steam generators (J B) 8 figs

MATERIALS

352
(IPEN-PUB-311)
Preparation of U_3O_8 powder for MTR type fuel from ammonium uranyl carbonate. Marcondes, G H , Riella, H G (Instituto de Pesquisas Energeticas e Nucleares (IPEN), Sao Paulo, SP (Brazil)) Aug 1990 24p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607096

In this paper it is described the research done at IPEN-CNEN/SP on the preparation of U_3O_8 powder from calcination of the AUC, with appropriate characteristics to be used as dispersoid for MTR type fuel The calcination in air of the AUC leads a U_3O_8 powder that is further processed to obtain a powder with density and particle size as especifications The important process parameters are here discussed with the variation AUC calcination tem perature and sintering time of the U_3O_8 powder (author)

353
(Juel Conf-71 pp 228 242)
Life fraction rules. Maile, K (Stuttgart Univ (Germany F R) Staatliche Materialpruefungsanstalt) Apr 1989 557p NTIS (US Sales Only), PC A24/MF A01 Order Number DE91717019 (CONF-890186-)

From Workshop on structural design criteria for HTR Julich (Germany, F R) (31 Jan 1 feb 1989)

In Proceedings of the workshop on structural design criteria for HTR

Evaluations for lifetime estimation of high temperature loaded HTR-components under creep fatigue load had been performed The evaluations were carried out on the basis of experimental data of strain controlled fatigue tests with respectively without hold times performed on material NiCr 22 Co 12 Mo (Inconel 617) Life prediction was made by means of the linear damage accumulation rule Due to the high temperatures no realistic estimates of creep damage can be obtained with this rule Therefore the rule was modified The modifications consist in a different analysis of the relaxation curve including different calculation of the creep damage estimate resp in an extended rule, taking into consideration the interaction between creep and fatigue In order to reach a better result transparency and to reduce data set dependent result scattering a round robin with a given data set was carried out The round robin yielded that for a given test temperature of T = 950deg C realistic estimate of damage can be obtained with each modification Furthermore a reduction of resulting scatterbands in the interaction diagram can be observed, i e the practicability of the rule has been increased (orig)

354
Research and development of fast breeder reactor, 8: Research and development on structures and materials. (Power Reactor and Nuclear Fuel Development Corp , Oarai, Ibaraki (Japan) Oarai Engineering Center) *Donen Giho (PNC Technical Review) (Japan),* No 73, 81-92 (Mar 1990) (In Japanese)

This paper summarizes the progress and the major achievements of R and D on structures and materials pertain ing to the development of JOYO' MONJU and future FBR plants At the stage of JOYO the design guide was established based on available relevant design criteria with findings typically about sodium technology incorporated At the stage of Monju , R and D for covering wider areas and developing more sophisticated methodology was attempted to meet the further requirements for the design of a prototype FBR Various types of material tests were conducted to develop the 'Material Strength Standard A general purpose nonlinear structural analysis system was developed, and structural component tests were carried out to verify associated design evaluation methods The elevated temperature structural design criteria were established based on these activities Further efforts are being made aiming at the extention and advancement of accumulated technology and the application of new materials and structures (author)

355
Chemical reactions of caesium, tellurium and oxygen with fast breeder reactor cladding alloys. Pt. 4: The corrosion of ferritic steels. Pulham R J , Richards, M W *Journal of Nuclear Materials, on Metallurgy, Ceramics and Solid State Physics in the Nuclear Energy Industry (Netherlands),* 172 No 3, 304-313 (Aug 1990)

A study of the corrosion of the steels FV448 and DT2203Y05 by Cs/Te mixtures in sealed capsules containing oxygen buffers at 948 K after 168 h shows that the oxide dispersion strengthened steel is more resistant to corrosion than is FV448 at Cs Te ratios of 1 1 and 2 1 Both steels generally corrode evenly and show more resistance to the more damaging intergranular penetration than do PE16 and M316 alloys Corrosion is most severe at 1 1 compositions irrespective of oxygen potential, and the corrosion products are Cs_2Te + transition metal tellurides The corrodants Cs_2Te, Cs_2Te+Cs and Cs are inert to FV448 in the absence of O_2 but corrosion increases with increasing oxygen potential At low potentials the dominant corrosion products are caesium chromates + Cr_2Te_3 and these are augmented by $CsFeO_2$ + $FeTe_{0\,9}$ at higher potentials The various types of corrosion are summarised (orig)

356
Electrochemical methods of determining the susceptibility to localized corrosion in hardfacing materials exposed to decontaminating solutions for a liquid metal fast breeder reactor. Eremias, B , Stefec, R *Werkstoffe und Korrosion (Germany, F R),* 41 No 9, 514-518 (Sep 1990)

Agents commonly used as decontaminating solutions for liquid metal cooled fast breeder reactors include nitric acid and potassium permanganate (solution No 1) in combination with oxalic acid (solution No 2) Overlayed or sprayed hardfacing alloys used for the valves and accessories of fast breeder reactors are less resistant to localized corrosion in these solutions and are generally inferior to AISI 316L stainless steel which is the substrate for the hardfacing The corrosion mechanism acting on these alloys when exposed to the No 1 and No 2 decontaminating solutions makes it possible to apply electrochemical techniques to ascertain the susceptibility to localized attack and to rapidly screen the degree of resistance in these environments (orig)

FUEL ELEMENTS

357
Release of volatile carbon-14 containing products from Zircaloy. Kopp, D , Muenzel, H *Journal of Nuclear Materials, on Metallurgy, Ceramics and Solid State Physics in the*

Nuclear Energy Industry (Netherlands), 173 No 1, 1-6 (Sep 1990)

Zircaloy is the standard material for fuel claddings in nuclear thermal reactors Due to its nitrogen content, carbon 14 (^{14}C) is produced during burn-up in a $^{14}N(n, p)^{14}C$ reaction For safety assessments the behavior of the ^{14}C under the proposed storage conditions has to be known In this contribution data on the release of volatile ^{14}C products from Zircaloy in the temperature interval from 200 to 600deg C in a pure argon atmosphere are given The influence of oxygen impurities in Zircaloy and in the argon atmosphere was also studied (orig)

358

Pellet-clad bonding during PCMI. Yu, A , Walker, S P , Fenner, R T *Nuclear Engineering and Design (Netherlands),* 121 No 1, 53-58 (Jul 1990)

Pellet-clad mechanical interaction adjacent to the mouths of radial cracks in the pellet can lead to local clad strain intensification In particular, if during operation bonding has occurred between the pellet and the clad, preventing relative azimuthal motion, very high strains would result and many clad failures would be expected These are however not observed We present here the results of a detailed analysis of the stress field in the region of the mouth of the pellet crack These show there to be very high radial tensile stresses not previously predicted, acting on the bond Such stresses provide a highly plausible mechanism for the failure of the bond These will permit the strain associated with the opening of pellet cracks during power ramps to be distributed over a relatively large arc of the cladding, and thus explain the low failure rate observed (orig)

359

Pellet-clad interaction (PCI) failures of zirconium alloy fuel cladding - a review. Cox, B *Journal of Nuclear Materials, on Metallurgy, Ceramics and Solid State Physics in the Nuclear Energy Industry (Netherlands),* 172 No 3, 249 292 (Aug 1990)

This review summarizes the history of the appearance and cure of pellet-cladding interaction (PCI) failures during the operation of Zircaloy clad UO_2 fuel in a number of reactors The work carried out to permit unrestricted operation of reactors without causing PCI failures has led to the universal adoption of the CANLUB-graphite coated cladding in CANDU reactors, and to the wide adoption of Zr-liner cladding in BWRs There has only been a low incidence of PCI failures in PWR cladding, and the problem has not loomed large enough to require the adoption of either of the above protective methods in these reactors, although experimental liner cladding has been tested The extensive work on the mechanism of PCI failures (leading to the conclusion that an SCC process induced by fission product iodine is the most probable cause) is summarised (orig)

360

System and method for removing and consolidation fuel rods of a nuclear fuel assembly. Ellingson, F J , Kapoor, A Kramer, A W , Sherwood, D G (to Westinghouse Electric Corp , Pittsburgh, PA (USA)) USA Patent 4,952,072/A/ 28 Aug 1990 Filed date 16 Feb 1989 vp Patent and Trademark Office, Box 9, Washington, DC 20232 (USA)

This patent describes a system for removing and consolidating the fuel rods of a fuel rod assembly The assembly has many grids, each of which includes an array of rod-receiving cells for receiving and retaining a fuel rod

361

Segment fuel element. Oe, Akira (to Nuclear Fuel Industries Ltd , Tokyo (Japan)) Japan Patent 2-138896/A/ 28 May 1990 Filed date 19 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

In a so called segment type fuel element comprising several segment fuels fastened by means of screws disposed at the end plug of fastening portions, a sectional portion with lower strength than that of a fuel can is disposed to the end plug of the fastening portion Thus when a tensile load is applied to the segment fuel in a fastened state, rupture is caused to the weakened cross sectional portion, that is, a narrow neck o the side of the main thread to protect the fuel can against rupture Accordingly, release of FP gases into the circumstance due to the rupture of the fuel can be inhibited thereby enabling to ensure the safety and serve the segment fuel to a rapid power-up stage in an experimental reactor (T M)

362

Method of heat treatment for reactor fuel can. Ito, Kunio (to Nippon Nuclear Fuel Development Co Ltd , Oarai, Ibaraki (Japan)) Japan Patent 2-136784/A/ 25 May 1990 Filed date 18 Nov 1988 3p (In Japanese) JAPIO Also available from INPADOC

A fuel can is constituted as a dual layer structure comprising a inner circumferential pure zirconium tube made of a pure zirconium and an outer circumferential zirconium base alloy tube made of a zircaloy layer, and as a final step of thermal treatment, it is annealed under conditions at a temperature of 650 - 675degC for 100 - 200 sec Since the pure inner zirconium layer is softened by such step, it can sufficiently moderate pellet-cladding interaction (PCI) caused upon burning in the reactor On the other hand, the outer zirconium alloy layer has a greater strength than that when annealed completely, and PCI resistance can rather be improved Accordingly, the zirconium liner clad tube developed for a longer life can be provided with optimum strength property (T M)

363

Support lattice for fuel assembly. Sakaguchi, Noriyuki (to Nippon Nuclear Fuel Development Co Ltd , Oarai, Ibaraki (Japan)) Japan Patent 2-136786/A/ 25 May 1990 Filed date 18 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

In a support lattice for a fuel assembly in which at least one of support members for fuel cans comprises a lantern spring made of inconel alloys, plating layers of beryllium or aluminum metals which are electrochemically less noble than the materials of the fuel can are formed to a thickness of 20 to 40 μm to the surface of the lantern spring abutting against the fuel can This can constitute a local cell, and reduce the amount of corrosion of the fuel can as less as possible to improve the corrosion resistance of the fuel can (T M)

364

Method of charging spent fuel. Ueda, Makoto (to Nippon Atomic Industry Group Co Ltd , Tokyo (Japan), Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-128200/A/ 16 May 1990 Filed date 8 Nov 1988 8p (In Japanese) JAPIO Also available from INPADOC

In the method of charging spent fuels in the present invention, fuels can be charged at a high volume efficiency while keeping critical safety That is, when the spent fuels are charged into a spent fuel transportation container or a spent fuel storage rack having many containing holes, the fuels are charged while monitoring neutron flux levels If there is a worry that the neutron flux level may exceed an allowable limit level the already charged spent fuel having the greatest neutron multiplication factor is withdrawn from the containing hole After the withdrawal, a non-fuel material is charged to the containing hole so that spent fuels can not more be charged Accordingly, the sub criticality of the spent fuels is ensured In this case it is preferable to use neutron absorbers for the non fuel material (I S)

CONTROL SYSTEMS

365
(AECS-SDB–12)
Executive summary of the guidebook on training to establish and maintain the qualification and competence of nuclear power plant operations personnel. (International Atomic Energy Agency, Vienna (Austria)) 1990 16p Translation of 'Executive summary of the guidebook on training to establish and maintain the qualification and competence of nuclear power plant operations personnel IAEA Vienna 1989 NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607829

Translation of 'Executive summary of the guidebook on training to establish and maintain the qualification and competence of nuclear power plant operations personnel' IAEA Vienna, 1989

Since the IAEA published its guidebook on the qualification of nuclear power plant operation personnel in 1984 (Technical Reports Series no 242) there have been important developments in the approach for training adopted by many operating organizations in different countries The guidebook, described in this report, proposes an approach which is comprehensive and systematic in its methodology and cost effective in its implementation 5 refs 1 fig

366
(BNL-NUREG–45316)
Detection and mitigating rod drive control system degradation in Westinghouse PWRs. Gunther, W Sullivan, K (Brookhaven National Lab , Upton, NY (USA)) [1990] Contract AC02-76CH00016 14p (CONF-9010220–7) NTIS, PC A03/MF A01, OSTI, INIS, GPO Dep Order Number DE91004017

From 1990 IEEE nuclear science symposium, Arlington, VA (USA) (23-27 Oct 1990)

A study of the effects of aging on the Westinghouse Control Rod Drive (CRD) System was performed as part of the US NRC s Nuclear Plant aging Research (NPAR) Program For the study, the CRD system boundary includes the power and logic cabinets associated with the manual control rod movement, and the control rod mechanism itself The aging related degradation of the interconnecting cables and connectors and the rod position indicating system also were considered This paper presents the results of that study pertaining to the electrical and instrumentation portions of the CRD system including ways to detect and mitigate system degradation

367
(EGG-EE–8935)
Neural network setpoint control of an advanced test reactor experiment loop simulation. Cordes, G A , Bryan, S R , Powell, R H , Chick, D R (EG and G Idaho, Inc , Idaho Falls, ID (USA)) Sep 1990 Contract AC07-76ID01570 63p NTIS, PC A04/MF A01 - OSTI, GPO Dep Order Number DE91001844

This report describes the design, implementation, and application of artificial neural networks to achieve temperature and flow rate control for a simulation of a typical experiment loop in the Advanced Test Reactor (ATR) located at the Idaho National Engineering Laboratory (INEL) The goal of the project was to research multivariate, nonlinear control using neural networks A loop simulation code was adapted for the project and used to create a training set and test the neural network controller for comparison with the existing loop controllers The results for three neural network designs are documented and compared with existing loop controller action The neural network was shown to be as accurate at loop control as the classical controllers in the operating region represented by the training set 9 refs , 28 figs , 2 tabs

368
(EPRI-MD–6881 Vol 2, pp 14 1-14 17)
CANDU computerized safety system. Popovic, J R , Hinton, G J (Atomic Energy of Canada Ltd , Mississauga, Ontario (Canada)) Jul 1990 422p Research Reports Center, Box 50490, Palo Alto, CA 94303 (CONF-8912108–Vol 2)

From Conference on advanced computer technology for the power industry, Scottsdale, AZ (USA) (4-6 Dec 1989)

In Proceedings 1989 conference on advanced computer technology for the power industry Volume 2, Computer technologies

In CANDU 6 PHW reactors, automatic reactor trip has been carried out with digital computers since 1982 The introduction of computers in the CANDU shutdown systems was motivated by the excellent operating experience with digital systems used for direct digital control of the CANDU generating stations and by new licensing requirements for enhanced plant CANDU 6 reliability This paper summarizes the evolution of digitally based design to meet these requirements, the system configuration of a fully computerized shutdown system and the associated design philosophy, software and hardware qualification process, and the operating experience of the existing installations The fully computerized Shutdown System is described for the Darlington Nuclear Generating Station (one unit reached criticality in November, 1989 and is scheduled to be in service in 1990) Future design developments are also discussed

369
(IPEN-PUB–298)
A failure detection and isolation system simulator. Assumpcao Filho, E O , Nakata, H (Instituto de Pesquisas Energeticas e Nucleares (IPEN), Sao Paulo, SP (Brazil)) Apr 1990 16p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607651

A failure detection and isolation system (FDI) simulation program has been developed for IBM-PC microcomputers The program, based on the sequential likelihood ratio testing method developed by A Wald, was implemented with the Monte Carlo technique The calculated failure detection rate was favorably compared against the wind-tunnel experimental redundant temperature sensors (author)

370
(JAERI M-90-104)
Temperature analysis of control rod for HTTR. Maruyama, So

Nishiguchi, Isoharu, Fujimoto, Nozomu, Shiozawa, Shusaku, Sudo, Yukio, Ogura, Takeshi (Japan Atomic Energy Research Inst , Tokyo (Japan)) Jul 1990 64p NTIS (US Sales Only), PC A04/MF A01 Order Number DE91723317

For the High Temperature Engineering Test Reactor, the reactor shut down from the high temperature condition is made by the two-step control rods insertion in order to avoid a fatigue damage of control rod casing at high temperature The control rods in reflec tor region are inserted into the core immediately after reactor scram and then, the control rods in fuel region are inserted when the core temperature becomes enough lower than the temperature limit of materials Control rods in fuel region are inserted automatically by either signal of prescribed time given by timers or the signal that reactor outlet coolant temperature reached the preset value This report describes the method, condition and results of control rod temperature analysis under various scram conditions (author)

371
(JINR E-10-90-323)
Clustering methods and visualization algorithms to aid nuclear reactor operative diagnostics. Pepyolyshev Y N , Dzwinel, W (Joint Inst for Nuclear Research, Dubna (USSR) Lab of Neutron Physics) [1990] 21p (CONF 900710-1) NTIS (US Sales Only), PC A03/MF A01, OSTI, JINR Publishing Department, Head Post Office, P O Box 79 101000 Moscow USSR Order Number DE91001200

From Balancing automation and human action in nuclear power plants, Munich (Germany F R) (9-13 Jul 1990)

Noise and vibration analysis appears to be a valuable tool for nondestructive nuclear reactor condition monitoring It enables early fault detection and gives the opportunity to avoid abnormal operation and accidental situations To make the efficient and convenient tool for an operator the software system should enable the proper signal processing speed, its handling must be as easy as possible and the maximum information should be monitoring in the form of comprehensive patterns, relatively simple to analyze to perform suitable action Pattern recognition techniques for noise analysis gives such an opportunity The pioneering proposition of clustering application for a reactor control are presented in [3,4] and still have been used and developed (e g [5,6]) The reliability of such an approach depends mainly on the proper choice of cluster ing algorithm In this paper some of pattern recognition methods for reactor operative diagnostics are recommended and the software system based on them is presented In the last but not the least chapter, the information obtained by an operator is exemplified The conclusions are reported at the end of the paper

372
(ORNL/FTR-3801)
[Nuclear reactor surveillance and diagnostics]. Kryter, R C (Oak Ridge National Lab , TN (USA)) 1 Nov 1990 Contract AC05-84OR21400 7p NTIS, PC A02/MF A01 - OSTI, GPO Dep Order Number DE91002803

The International Program Committee for SMORN VI met in Paris to organize the contributed technical papers into sessions and to appoint session chairmen These objectives were accomplished without controversy SMORN VI has the makings of successful venture, with 80 papers submitted by authors from 20 countries throughout Europe, Asia, and the Western Hemisphere For the first time in this symposium series, papers were received from Cuba and the USSR

373
Elektrische Antriebe des Sicherheitssystems von Kernkraftwerken (Electrical drives of the safety system in nuclear power plants). Berlin (Germany, F R), Beuth (Sep 1990) 27p (In German) (DIN-44834)

Actuating drives, control magnets for ventilators, machine drives and control member drives are part of this rule The rule deals with the security and technical requirements for design, construction, calculation, fabrication, assembling, testing and operation Furthermore, it places significant demands, with regard to planning and arrangement of electrical drives, on the accompanying technical systems Furthermore, demands are placed on the aggregate protection for electrical drives of the security systems The signals given to these systems do not, however, have precedence over the protection signals of the reactor The rule is identical with KTA-3504, version 9/1988 (orig./HP)

374
In-plant emergency protection. Birkhofer, A pp 41-61 of 8th German symposium on atomic energy law Proceedings Lukes, R , Birkhofer, A (eds) Koeln (Germany, F R), Heymanns (1989) 288p (In German) (CONF-8903251-)

From 8 German symposium on atomic energy law, Muenchen (Germany, F R) (1-3 Mar 1989)

In-plant emergency protection complements preventive and precautionary risk provisions It is meant as a precaution against extremely improbable beyond-design events and comprises both preventive and limiting measures Priority is given to damage prevention In-plant emergency protection is basically different from safety design due to the varying technologies involved However, this difference or 'diversity' is the great strength of this fourth safety net Diversity in engineered safety is as much an effective means to protect against hidden systematic defects of redundant systems as diversity is in the case of in-plant emergency protection on the level of safety strategy since it will provide for a compensation of systematic weakness in safety design which cannot be precluded (orig / HSCH)

375
Computers replaced at Finland's Loviisa PWR - on-line and on-time. Manninen, Teemu *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 23-24, 26 (Jul 1990)

Replacement of the Loviisa PWR's three process computer systems - among the largest and most advanced in the world -was completed on schedule earlier this year after a complex and intensive installation programme stretching over three years Sophisticated techniques and meticulous planning, going back to 1984, allowed the new systems to be connected and tested alongside the old without interfering with their running, the final exchange being done during normal plant operation (author)

376
Upgrade, rebuild or replace?. Forbes, C A. *Nuclear Engineering International (Incorporates Nuclear Power) (UK);* 35 No 432, 43-44 (Jul 1990)

Ageing reactor simulators present some tough decisions for utility managers Although most utilities have chosen the cheaper upgrading solution as the best compromise between costs and outage length, some US utilities have found that for them, replacement represents the best option Simulators may be less than ten years old, but they have limited instructor systems, older low fidelity models that cannot reproduce important training scenarios, and out of date, difficult to maintain computers that do not permit much expansion of the models anyway Perhaps worse than this is the possibility that the simulator may no longer be a faithful reproduction of the referenced plant, or have poor (or nonexistent) documentation (author)

377
System software for solving state monitoring tasks in nuclear power plants. Papp, Jozsef, Tothmatyas, Istvan *Mueszeruegyi es Merestechnikai Koezlemenyek (Hungary),* 26 No 48 23 27 (1990) (In Hungarian)

Acoustic and vibration monitoring systems have been developed for some reactor diagnostics and supervisory tasks of the Paks Nuclear Power Plant The basic software of this monitoring system is described The DIR-90 software consists of several programs controlling various tasks and functions of the monitoring system This software can be interfaced with an expert shell system (R P) 2 figs

378
Using the expert knowledge in online control systems for NPP units. Gorlin, A I , Dmitriev, V M , Kroshilin, A E , Lesnoj, A E *Ehlektricheskie Stantsii (USSR),* No 12, 13-18 (Dec 1989) (In Russian)

Peculiarities of expert systems presenting a new generation of on-line control system for NPP units are considered The expert systems comprise software and hardware which use expert knowledge duty formalized for problem solution and to a certain extent they are modelling the work of an expert The PEhKS simple expert carcass system operating in regime of adviser for different problems of operation of units and separate components of equipment and EhDEhS express-diagnostic system for identification of emergency regimes are described

379
On the survivability of NPP technological on-line control systems under earthquakes. Litinskij, G I , Yastrebnetsij, M A *Teploehnergetika (USSR),* No 10, 30-35 (Oct 1989) (In Russian)

The purpose of the paper is to formulate principles of providing survivability of NPP technological process on line control systems under seismic effects, that is principles of providing performance of the system with assigned quality indices under earthquakes and after them under all conditions of NPP failures of components of technological process on line control systems, methods for forcasting reliability and main reliability and main methods for improving system reliability under seismic effects are analyzed It is proved that fulfilment of seismic stability requirements dosen t lead to sufficient rise in the system price, because the most expensive measures are required for other types of accidents with main NPP equipment

380
Computer-aided design of components for information and calculation systems of TPPs and NPPs. Vishnyakov, L N , Rudnev, V E , Shapiro, V I *Teploehnergetika (USSR),* No 10, 35-39 (Oct 1989) (In Russian)

The main problems, solved when creating computer-aided design systems (CADS) for components of information and calculation systems for TPPs and NPPs are considered It is noted that principle direction of CADS technology development lies in elongation of continuous chain of the problems Another important direction is organization and accumulation of computerized bank of standard design solutions Application of personal computers in CADS is considered to be very perspective

381
Online determination of power distribution within reactor fuel pins. Kalya, Zoltan, Pos, Istvan, Cserhati, Csaba *PAV Koezlemenyek (Hungary),* No 2, 44-48 (1989) (In Hungarian)

A new calculation method based on hot spot monitoring has been developed at Paks Nuclear Power Plant for reactor power limitation The calculations can be divided into preliminary and online real-time computations so that the final results remain unchanged and the online part can be made extremely fast The preliminary calculations are performed by a dedicated program for solving thermal diffusion equations by finite element method inside the fuel assembly The algorithm for online calculation utilizes neutron flux and temperature detector data for the determination of actual reactor state values (R P) 4 refs

382
Design and checking of a large ADA real-time system. Theron, J L *Nuclear Instruments and Methods in Physics Research, Section A (Netherlands),* 293 No 1/2, 373 376 (1 Aug 1990) (CONF-891094–)

From International conference on accelerator and large experimental physics control systems (ICALEPCS), Vancouver (Canada) (30 Oct - 3 nov 1989)

SEMA Group, a European software company, is currently developing the control system for France's 1400 MW nuclear power plants The project is described briefly in terms of user needs and technical challenges The technical choices are also described, and currently available statistics on the project are given (orig)

383
Development of a PWR CRDM [control rod drive mechanism] data-analyzing system. Miyaguchi, Jinichi *Transactions of the American Nuclear Society (USA),* 60 466-468 (1989) (CONF-891103–)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Control rod drive mechanisms (CRDMs) play an important role in the nuclear power plant, and their reliability impacts plant operation and reactor safety The CRDM performance might decline if the CRDM has been operated for a long time The CRDM's operation time is expected to increase significantly, depending on the variations of plant operation, so it is desirable to upgrade preventive maintenance of CRDMs and drive lines through periodic inspection and condition monitoring Furthermore, in the case of CRDM malfunction, it is necessary to cope immediately with the trouble, based on technical judgment The CRDM data-analyzing system has been developed in order to achieve highly reliable CRDMs by predicting malfunctions

384
Force analysis of the advanced neutron source control rod drive latch mechanism. Damiano, B *IEEE*

Transactions on Nuclear Science (Institute of Electrical and Electronics Engineers) (USA), 37 No 3, 1415-1423 (Jun 1990) (CONF-900143-)

From Institute for Electronic and Electrical Engineers (IEEE) nuclear science symposium, San Francisco, CA (USA) (15-19 Jan 1990)

The advanced neutron source reactor (ANS), a proposed Department of Energy research reactor currently undergoing conceptual design at the Oak Ridge National Laboratory (ORNL), will generate a thermal neutron flux approximating 10^{20} M^{-2} - S^{-1} The compact core necessary to produce this flux provides little space for the shim/safety control rods, which are located in the central annulus of the core Without proper control rod drive design, the control rod drive magnets (which hold the control rod latch in a ready-to-scram position) may be unable to support the required load due to their restricted size This paper describes the force analysis performed on the control rod latch mechanism to determine the fraction of control rod weight transferred to the drive magnet

385

Nuclear excitation laser type incore neutron measuring system. Naka mua Hisashi, Nakazawa Masaharu (to Power Reactor and Nuclear Fuel Development Corp , Tokyo (Japan)) Japan Patent 2 138899/A/ 28 May 1990 Filed date 18 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC

In a nuclear excitation laser type incore neutron measuring system according to the present invention, a laser oscillator filled with nuclear exci taion gases such as ^3He, KrF or XeF is disposed to the top end of the control rod Then, since the top end is situated to the reactor core upon pulling up, the control rod, the neutron excitation gases are excited into plasmas by means of neutrons or nuclear fission products The nuclear excitation gases excited into plasmas emit laser beams by themselves Otherwise, they amplify the laser beams entering from the outside These optical responses are introduced into an optical processing system disposed at the outside and the neutron energy is discriminated based on the spectrum Accordingly, neutron behaviors are monitored extremely finely by measuring the neutron density and neutron flux on every energy The detection sensitivity can be improved by forming reactive membranes such as U_3O_8 in the laser device (N H)

386

Control rod drive. Kaneko, Tadashi (to Toshiba Corp , Kawasaki, Kanagawa (Japan), Nippon Atomic Industry Group Co Ltd , Tokyo (Japan)) Japan Patent 2-136787/A/ 25 May 1990 Filed date 17 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

If a pin and a roller made of a cobalt-based alloy are used in electrically actuating control rod drives, cobalt is incorporated into abrasion products, which are brought into a reactor by means of coolants to increase the radiation doses In view of the above, the substrate for the pin is made of an iron-based alloy and a nitride layer is formed at least on the substrate at the surface of the pin suffering from sliding abrasion in contact with the surface of a hole of the roller and a ceramic layer is coated over the nitride layer Since the ceramics are liable to be tipped, the substrate of the pin is made of metal and the iron-based alloy is selected in view of the easiness for the nitridation and excellent wear resistance in water Further, for improving the corrosion resistance of the nitride layer in a purified water circumstance at high temperature, it is coated with ceramic layer Thus, frictional abrasion between the pin and the roller accompanying the start-up and shutdown, as well as increase and decrease of power in the reactor can be reduced and operator's radiation exposure can be reduced (N H)

387

Travelling in-core probe. Hirose, Jun (to Toshiba Corp , Kawasaki, Kanagawa (Japan), Nippon Atomic Industry Group Co Ltd , Tokyo (Japan)) Japan Patent 2-129594/A/ 17 May 1990 Filed date 10 Nov 1988 5p (In Japanese) JAPIO Also available from INPADOC

An object of the present invention is to obtain a travelling in-core probe (TIP) capable of detecting a integrity of a detector and positional detection for vertical direction (Z direction) by a computer The TIP comprises a detector movable vertically in a reactor core A plurality of detector of the same kind are disposed vertically adjacent with each other at the end of cable that sends detection signals from the TIP A difference between the output from the plurality of detectors is obtained by the thus constituted TIP Then, since fluctuations of the neutron fluxes are removed, the recesses in the outputs due to fuel spacers, etc are made distinct, enabling to detect the position in the Z direction Further, the integrity of the detector can be judged by determining each of the deviations of a plurality of detectors (I S)

388

Method of exchanging cables of neutron monitoring instrumentation tube and folding device of the cable. Sakamaki, Kazuo (to Toshiba Corp , Kawasaki, Kanagawa (Japan)) Japan Patent 2-128193/A/ 16 May 1990 Filed date 7 Nov 1988 6p (In Japanese) JAPIO Also available from INPADOC

In a BWR type reactor, a wide range monitor (WRNM) is used instead of a conventional neutron source range monitor (SRM) or an intermediate range monitor (IRM) The WRNM is always fixed to a predetermined position in a reactor core while containing a detection section in a dry tube, different from a conventional monitor Accordingly, driving devices for the conventional detection section such as in SRM and IRM are not necessary but, when the reactor is operated for a long period of time, it is sometimes necessary to be replaced with new WRNM According to the present invention, the cable of the detector placed in a neutron instrumentation tube is connected to a cable take-up drum in a take-up device passing through a cask Then, the cable is taken-up by driving the take-up drum by a driving motor and the WRNM detection section attached to the top end of the cable is contained in the cask With this constitution, replacing and processing operation for the detection section can be facilitated and operator's exposure dose can be reduced (I S)

389

Recycling pump control device. Suzuki, Toshiyuki, Kawasaki, Takahide (to Hitachi Ltd , Tokyo (Japan)) Japan Patent 2-126197/A/ 15 May 1990 Filed date 7 Nov 1988 4p (In Japanese) JAPIO Also available from INPADOC.

The device of the present invention is suitable for ensuring thermal margin of a nuclear reactor upon failure of control rod insertion That is, a recycling pump is tripped only when reactor scram signals are generated and neutron fluxes in the reactor are decreased In the process of the present invention, lowering of water level in the steam drum or the reactor is suppressed by tripping or decelerating the recycling pump According to the present invention, when the control rods are inserted certainly to the reactor core upon reactor scram, lowering of the water level in the steam drum can be suppressed by tripping the recycling pump Further, upon failure of control rods insertion since the recycling pump is kept operated to cool the reactor the thermal margin of the reactor can be ensured (I S)

390
Power control device for BWR type reactor. Takigawa Yukio Ebata Shigeo (to Nippon Atomic Industry Group Co Ltd Tokyo (Japan) Toshiba Corp Kawasaki Kanagawa (Japan)) Japan Patent 2-114199/A/ 26 Apr 1990 Filed date 25 Oct 1988 4p (In Japanese) JAPIO Also available from INPADOC

The power control device for a BWR type reactor according to this invention can prevent power change and keep stable operation state when abnormality such as load interruption of electric generator or turbine trip, etc occurs That is, the power control device comprises a recycling pump for compalsorily circulating reactor coolants to control the reactor core power, a detector for detecting the open/close state of turbine main steam check valve and a speed limiter for outputting a run-back signal for decelerating the recycling pump to the lowest allowable flow rate upon receiving the valve closure signal from the detector With such a constitution if abnormality such as load interruption of electric generator or turbine trip should occur to rapidly close the turbine main steam check valve the abnormality is detected and the recycling pump is decelerated to a predetermined number of rotation As a result the reactor core power is controlled The reactor core flow rate corresponding to the number of rotation is greater than the spontaneous recycling flow rate (I S)

ENVIRONMENTAL ASPECTS

391
(BARC–1485)
Magnitudes and frequencies of earthquakes in relation to seismic risk. Sharma, R D (Bhabha Atomic Research Centre, Bombay (India)) 1989 40p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607178

Estimating the frequencies of occurrence of earthquakes of different magnitudes on a regional basis is an important task in estimating seismic risk at a construction site Analysis of global earthquake data provides an insight into the magnitudes frequency relationship in a statistical manner It turns out that whereas a linear relationship between the logarithm of earthquake occurrence rates and the corresponding earthquake magnitudes fits well in the magnitude range between 5 and 7, a second degree polynomial in M, the earthquake magnitude provides a better description of the frequencies of earthquakes in a much wider range of magnitudes It may be possible to adopt magnitude frequency relation for regions, for which adequate earthquake data are not available, to carry out seismic risk calculations (author) 32 refs , 8 tabs , 7 figs

392
(EGG-M–89357)
Mechanical design and fabrication of a prototype facility for processing NaK using a chlorine reaction method. Dafoe, R , Keller, D , Stoll, F (EG and G Idaho, Inc , Idaho Falls, ID (USA)) [1990] Contract AC07-76ID01570 7p (CONF-900210–54) NTIS, PC A02/MF A01 OSTI, GPO Dep Order Number DE91001849

From Waste management 90 working towards a cleaner environment waste processing transportation storage and disposal technical programs and public education Tucson AZ (USA) (25 Feb - 1 mar 1990)

A prototype facility has been built at the Idaho National Engineering Laboratory (INEL) to dispose of 180 gal(0 68 m^3) of radioactively contaminated NaK (sodium-potassium) that have been stored on site for 35 years The NaK was used as primary coolant for the Experimental Breeder Reactor I (EBR-I) at the INEL and was contaminated during a meltdown of the Mark II core in November 1955 The NaK then was transferred to four containers for temporary storage The facility process will react the NaK with elemental chlorine using a batch process to produce chemically stable sodium chloride and potassium chloride salts The first use of the facility will be on a prototype level to verify the method If results are favorable, the facility will be modified to eventually dispose of the EBR-I NaK The design and intended operation of the prototype facility are described 2 figs

393
(IPEN-PUB–305)
Evaluation of the population dose due to the gaseous emission of a radioisotopes production unit. Gordon, A M P L , Jacomino, V M F , Sordi, G -M A A (Instituto de Pesquisas Energeticas e Nucleares (IPEN), Sao Paulo, SP (Brazil)) May 1990 44p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607396

In order to control the emission of gaseous radioactive iodine from the unit responsible for the production of radioisotopes of IPEN-CNEN/SP, a discharge monitoring is carried out In 1988 an activity of 65 GBq of I-131 was discharged to the environment Based upon this value and the site analysis, the effective equivalent dose in the general public was evaluated for normal operation and for an incidental discharge The evaluation was carried out by using a diffusion atmospheric model, 500 to 7000 m away from the discharge point and using 8 different wind direction sectors The critical group was identified as being the people who lives 3000 m far from the discharge point, in the diffusion sector NW The dose evaluated at this point is 10^9 times lower than the annual dose limit for individual of the public, according to Radiological Protection Standards The derived limit for discharge of iodine was also evaluated and it was concluded that the IPEN-CNEN/SP can increase their production up to a level which results in an annual discharge of 1,5 x 10^{12} of I 131 (author)

394
(SV UK–1990-20)
Operative meteorological data base in Forsmark. Appelgren, A Hallberg, B Nordlinder, S (Swedish State Power Board, Vaellingby (Sweden)) 24 Jan 1990 37p NTIS (US Sales Only), PC A03/MF A01, OSTI, INIS Order Number DE91607400

This report describes how data collected during a field measurement campaign were analysed and compiled to create a data base for operative use The data base gives information about wind and atmospheric stability at five locations around the Forsmark nuclear power plant Sodar systems and a 100 m high tower at Forsmark were used

Temperature, wind speed and wind direction were measured by sensors on the tower, while wind speed and direction, and the standard deviation of the vertical wind, were monitored by the sodar systems This gave meteorological data from several heights At Forsmark, the temperature difference and the wind speed from the tower were used to determine the atmospheric stability At the sodar locations, the stability was deduced by employing a scheme which considered the season the time of day, the wind direction and the wind speed To create the operative data base, the wind speeds and wind directions, respectively, from two locations at the time were correlated A code for graphical and numerical presentation of the data from the data base was developed A special system of warnings was included featuring notification about phenomena such as sea breeze warnings about large variation in the wind conditions within the area and warnings for situations in which the meteorological conditions make the results from the atmospheric dispersion calculations uncertain This feature was implemented to alert the user to the fact that ordinary dispersion and dose calculations using meteorological data from a single point might give erroneous results The operative data base and the presentation code were integrated with the dispersion and dose calculation code AIRPAC/EMMA, which is to be used in case of increased releases from nuclear power plants The possibility to use the data from the operative data base in the dispersion calculations was investigated (Abstract Truncated)

395

Ueberwachung der Ableitung gasfoermiger und aerosolgebundener radioaktiver Stoffe. T. 1. Ueberwachung der Ableitung radioaktiver Stoffe mit der Kaminfortluft bei bestimmungsgemaessem Betrieb (Measuring and monitoring of gaseous releases and aerosol-bound radioactive material. Pt. 1. Measuring and monitoring of radioactive material released together with the stack exhaust air in regular operation). Koeln (Germany F R) Heymanns (1990) 27p (In German) (KTA-1503 1(draft ed 6/90))

This part of the standard is to be applied to devices for stationary nuclear power plants with LWR s and HTR s during regular operation Their duties are (a) balancing of the radioactive material released with the stack exhaust air as a basis for the assessment of the radiological consequences, (b) automatic release of signals if limits are exceeded, (c) contributing to meeting the requirements of paragraph 46, section 1, number 3 of the Radiation Protection Ordinance (orig)

396

Dungeness bibliography. Riley, H Research and Survey in Nature Conservation Peterborough (UK), Nature Conservancy Council (1989) 42p the British Library Document Supply Centre, Boston Spa, Wetherby, West Yorks LS23 7BQ

This report is a bibliography of material relating to the ecology of Dungeness in Kent and has been produced, under contract, for the Nature Conservancy Council Since its first appearance in 1985, much new material has been published or has come to light and is included in this 1989 edition The area concerned comprises the system of shingle beaches between Jury s Gap and Greatstone-on-Sea on the south Kent/Sussex coast, and is a site of international conservation importance It is one of the finest examples of an apposition shingle beach in the world and in terms of physiography, botany, entomology and ornithology it is an area of high scientific interest As well as being of great scientific importance, Dungeness is subject to a high degree of 'human' pressure excavation and building operations, military activity, vehicular damage and trampling threaten to destroy what is left of the natural shingle ecosystem at Dungeness In recognition of this the Nature Conservancy Council is supporting a renewed programme of research into the ecology of the site, to try to determine how best to protect the remaining intact areas (author)

397

Results of radioactivity survey accompanying port call of nuclear warships of US Navy (for December, 1989). (Science and Technology Agency Tokyo (Japan)) *Genshiryoku Anzen Iinkai Geppo (Japan),* No 135 21 (Mar 1990) (In Japanese)

The results of radioactivity survey accompanying the call at Yokosuka Port of the nuclear submarines of US Navy are as follows The Puffer entered Yokosuka Port on December 27, 1989 and departed on January 6, 1990 Monitoring boats surveyed on the day before the entry, on the day of the entry, during the stay and on the day of the departure along respective survey courses The surveyors carried out the continuous monitoring using the automatic alarm system at the headquarters of radioactivity countermeasures, and periodically patrolled the monitoring posts As the results, abnormality was not found at all (K I)

398

Main approaches to the selection criteria of sites for nuclear power complexes. Borovikova, N M , Nagovitsyna, L I , Karachev, I I , Tkachenko, N V , Novikova, N K , Los', I P *Gigiena i Sanitariya (USSR),* No 2, 46-47 (Feb 1990) (In Russian)

The problem on NPP site selection in the Ukranian SSR with regard to high density of power and industrial plants providing presevation of hygienic qualities of the environment, rational water use, radiation safety of population living in the NPP region is considered 32 criteria on 9 indices are suggested such as radiation exposure, epidemiologic state of the region, biota contamination with radioactive wastes earth alienation, seismic safety, floading, recreation worth, a set of adverse conditions, sanitary-hygienic state of the water reservoir Three stages of the system to work with criteria are suggested 'Rejecting criteria were used at the first stage, then the selected territory is analysed using 'comparison' criteria, and the third stage is the evaluation of the territory by experts

399

***Nitrogen and amino acids content* in lake Drukshyaj plankton organisms biocenoses grown in model experiments.** Krevsh, A V , Budrene, S F , Yankyavichyus, K K *Lietuvos TSR Mokslu Akademijos Darbai, Serija B Chemija, Technika, Fizine Geografija (USSR),* No 4, 3-9 (1989) (In Russian)

Biocenoses growth in lake Drukshyaj (from 1984 water reservoir of the Ignalina NPP) collected in July 1985 and grown in 2 various in composition culture media in medium close in composition of main minerals to water of high capacity reservoir (medium 1) and in medium Fitzjarld (medium 2), has shown that the medium affects the component composition of plankton, as well as dominating types of algae Phytoplankton was dominating component in biomass in both media In medium 1 dominate green algae and diatoms, in medium 2 - blue green algae Content of proteins and amino acids in biomass changed depending on duration of biocenoses growth when dominating green and diatoms in biocenoses mass grown in medium 1 it reached maximum on the 15th day, and when dominating blue-green algae in biocenoses biomass grown in medium 2 on the 30th day

400
Operational radiation protection inspection of radioactive emission and the environment of the Paks Nuclear Power Plant: Measurement results and conclusions (1983-87). German, Endre, Ormai, Peter, Rosa, Geza *PAV Koezlemenyek (Hungary),* No 2, 2-18 (1989) (In Hungarian)

Radionuclide emission data and environmental inspection results for the period 1983-87 are compiled and analyzed It was found that the PNPP has maintained all regulations Annual total emission has been kept well below the relevant limits except for ^{3}H release with effluents that reached 40-85% of the limit In environmental samples only few instances were radionuclides from PNPP In the period considered the excess annual radiation burden to the population from PNPP was in the order of 10^{-7} Sv effective dose equivalent less than one ten thousandth of natural radiation burden so that it can be neglected from radiation health risk aspects (R P) 24 refs 8 figs 4 tabs

REGULATION AND LICENSING

401
(NUREG-0040 Vol 14-No 2)
Licensee contractor and vendor inspection status report. (Nuclear Regulatory Commission, Washington, DC (USA) Div of Reactor Inspection and Safeguards) Oct 1990 111p NTIS, PC A06/MF A01 - GPO, OSTI, INIS

A fundamental premise of the Nuclear Regulatory Commission's (NRC) licensing and inspection program is that licensees are responsible for the proper construction and safe and efficient operation of their nuclear power plants The total government industry system for the inspection of commercial nuclear facilities has been designed to provide for multiple levels of inspection and verification Licensees contractors and vendors each participate in a quality verification process in compliance with requirements prescribed by the NRC's rules and regulations (Title 10 Code of Federal Regulations) The NRC performs an overview of the commercial nuclear industry by inspection to determine whether its requirements are being met by licensees and their contractors while the major inspection effort is performed by the industry within the framework of ongoing quality verification programs

402
(NUREG-0040-Vol 14-No 3)
Licensee contractor and vendor inspection status report. (Nuclear Regulatory Commission, Washington, DC (USA) Div of Reactor Inspection and Safeguards) Nov 1990 213p NTIS, PC A10/MF A01 - GPO, OSTI, INIS

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organization during the period from April 1990 through June 1990

403
(NUREG-0540-Vol 12-No 8)
Title list of documents made publicly available, August 1-31, 1990. (Nuclear Regulatory Commission, Washington DC (USA) Div of Freedom of Information and Publications Services) Oct 1990 314p NTIS, PC A14/MF A01 - GPO OSTI INIS

This monthly publication contains descriptions of the information received and generated by the US Nuclear Regulatory Commission (NRC) This information includes (1) docketed material associated with civilian nuclear power plants and other uses of radioactive materials and (2) nondocketed material received and generated by NRC pertinent to its role as a regulatory agency As used here, docketed does not refer to Court dockets, it refers to the system by which NRC maintains its regulatory records This series of documents is indexed by a Personal Author Index, a Corporate Source Index, and a Report Number Index The docketed information contained in the Title List includes the information formerly issued through the Department of Energy publication Power Reactor Docket Information, last published in January 1979

404
(NUREG-0750-Vol 32-No 2)
Nuclear Regulatory Commission issuances, August 1990. (Nuclear Regulatory Commission, Washington, DC (USA) Div of Freedom of Information and Publications Services) Aug 1990 79p NTIS, PC A05/MF A01 - GPO, OSTI, INIS

This report includes the issuances received during the specified period from the Commission (CLI), the Atomic Safety and Licensing Appeal Boards (ALAB), the Atomic Safety and Licensing Boards (LBP), the Administrative Law Judge (ALJ), the Directors' Decisions (DD), and the Denials of Petitions for Rulemaking (DPRM)

405
(NUREG-0750-Vol 32-No 3)
Nuclear Regulatory Commission issuances. (Nuclear Regulatory Commission, Washington, DC (USA) Div of Freedom of Information and Publications Services) Sep 1990 79p NTIS, PC A05/MF A01 - GPO, OSTI, INIS

This report includes the issuances received during the specified period from the Commission (CLI), the Atomic Safety and Licensing Appeal Boards (ALAB), the Atomic Safety and Licensing Boards (LBP), the Administrative Law Judges (ALJ), the Directors' Decisions (DD), and the Denials of Petitions for Rulemaking (DPRM)

406
(NUREG-0936-Vol 9 No 3)
NRC regulatory agenda. (Nuclear Regulatory Commission, Washington, DC (USA) Div of Freedom of Information and Publications Services) Oct 1990 139p NTIS, PC A07/MF A01 - GPO, OSTI, INIS

The Regulatory Agenda is a quarterly compilation of all rules on which the NRC has recently completed action or has proposed, or is considering action and of all petitions for rulemaking that the NRC has received that are pending disposition

407
(NUREG/CR-2000-Vol 9-No 9)
Licensee Event Report (LER) compilation for month of September 1990. (Nuclear Regulatory Commission, Washington, DC (USA) Office for Analysis and Evaluation of Operational Data, Oak Ridge National Lab , TN (USA)) Oct 1990 Contract AC05-84OR21400 106p (ORNL/NSIC-200-Vol 9-No 9) NTIS, PC A06/MF A01 - GPO, OSTI, INIS

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document The LERs, from which this information is derived are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1 16 and NUREG 0161 Instructions for Prepara tion of Data Entry Sheets for Licensee Event Reports For those events occur ring on and after January 1 1984 LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50 73 of the Code of Federal Regulations (10 CFR 50 73 – Licensee Event Report System) which was published in the Federal Register (Vol 48 No 144) on July 26, 1983 NUREG-1022, Licensee Event Report System – Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility Component, system, keyword and component vendor indexes follow the summaries Vendors are those identified by the utility when the LER form is initiated, the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System

408

Legal issues relative to in-plant emergency protection. Lukes, R pp 63-79 of 8th German symposium on atomic energy law Proceedings Lukes R Birkhofer A (eds) Koeln (Germany F R), Heymanns (1989) 288p (In German) (CONF-8903251–)

From 8 German symposium on atomic energy law, Muenchen (Germany, F R) (1 3 Mar 1989)

The paper discusses the legal issues and problems involved, gives an overview on the pertinent legal provision (sect 7 (2), 3 AtG) and outlines the associated measures to be assigned under aspects of completion stage (a) plant being planned, but not yet licenced, (b) plant under construction, (c) plant finished and in operation The assignment of measures to either the field of risk or the field of precaution will depend solely from the degree of probability of the occurrence of such a beyond-design event The outcome of such an assignment will also have consequences for the participation of the general public, for third-party protection and thus, for legal actions (orig /HSCH)

409

The current practice of the courts in nuclear facility licensing procedures. Rengeling, H W pp 291 305 of Mining law and energy law in the context of today s most urgent problems Festschrift for Fritz Fabricius on the occasion of his 70th anniversary Hueffer U Ipsen K Tettinger P J Stuttgart (Germany F R), Boorberg (1989) 516p (In German)

Under the heading of legal issues of the energy industry the author dis cusses court rulings bearing on the following issues of the nuclear facility li cencing procedure interpretation of the term facility', the obligatory structure of sect 7 (2,3) AtG, the contents of damage prevention in the atomic energy law, the staged licencing procedure espec the objects of partial licencings and the preliminary ruling, procedural issues and issues of judicial control (RST)

410

Core meltdown accidents belong into the category of remaining risks to be accepted. Decision of the Higher Administrative Court of Lueneburg, September 16, 1988. *Recht der Elektrizitaetswirtschaft (Germany, F R),* 51 No 3, 61-62 (Mar 1989) (In German)

Radioactive emissions due to incidents or accidents are not to be considered as events adding to an existing level of contamination of a site within the purview of section 45 StrlSchV (Radiation Protection Ordinance) Core meltdown accidents in the light of the law are to be assigned to the category of remaining risks to be accepted Headnotes of a decision by the Lueneburg Higher Administrative Court, - 7 OVG D 10/86 and 4/87 -, of September 16, 1988 (orig)

411

Prerequisites of a revocation of an operating licence for a nuclear power plant. Decision of the Higher Administrative Court of Muenster, December 19, 1988. *Recht der Elektrizitaetswirtschaft (Germany, F R),* 51 No 3, 62-64 (Mar 1989) (In German)

The headnotes of the court decision read as follows There is no legal basis supporting the demand for a general nuclear power phaseout There is a defined relation between an optional revocation according to section 17, sub-sec (3), no 2 AtG (Atomic Energy Act), and the obligatory revocation according to section 17, sub-sec (5) AtG Application of section 17, sub-sec (5) AtG requires proof of the fact that operation of the nuclear power plant involves an imminent danger to the group of persons defined in that sec tion Higher Administrative Court of Muenster, decision 21 AK 8/88 of De cember 19 1988 (orig)

412

Standardization in nuclear power. Janik, L *Nukleon (Czechoslovakia),* No 4, 14-17 (1989) (In Czech)

English translation available from Nu clear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

A brief survey of the preparation of standardization documents in advanced industrial countries is given The present state of issuing standards in Czechoslovakia and other CMEA countries is described and the activity in this field reviewed with respect to the use of standards in international relations and in Czechoslovakia (author) 2 figs , 1 ref

413

Regulatory perspective on seismic PRAs and margin studies. Vollmer, R H , Shao, L C , Chokshi, N C , Thadani, A C *Res Mechanica (UK),* 30 No 4, 377-387 (1990) (CONF-870812–)

From 9 biennial international conference on structural mechanics in reactor technology (SMIRT-9), Lausanne (Switzerland) (17-21 Aug 1987)

This paper describes the results of several seismic probabilistic risk assessments (PRAs) and margin studies, the engineering insights gained form them, and the impact of these studies on the regulatory decision-making process Two case studies are discussed demonstrating considerations of deterministic and probabilistic concepts in the licensing process Further possible regulatory uses of PRAs and margin studies are also discussed (author)

414
A critical review of nuclear power plant decommissioning planning studies. Lough, W T , White, K P Jr *Energy Policy (UK),* 18 No 5, 471-479 (Jun 1990)

During the past decade there have been at least ten major efforts to perform comprehensive, analytical studies of the complex issues associated with decommissioning civilian nuclear power plants These planning efforts are reviewed, using the standard framework of technology assessment In particular, each study is analysed to determine the degree to which formal methods of decision analysis have been employed to evaluate options and make recommendations and the degree to which formal methods of consensus have been employed to engage citizen involvement and promote public acceptance Not unexpectedly we find that the greatest strides in decommissioning analyses have been made in forecasting the economic costs of decommissioning to licensees Comparatively few improvements have been made in the processes used to compare the impacts of alternative technologies more broadly, or to address the legitimate concerns of interested parties more widely (author)

415
Configuration management: Are you operating the plant you licensed?. Smith, P R *Transactions of the American Nuclear Society (USA),* 60 449 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

Before a nuclear power plant becomes operational, its plant systems and structures are functionally arranged to operate in a particular way This functional arrangement is specified by the plant design requirements and is called its configuration This paper presents a method by which all plants (those under construction, as well as those in operation) can benefit from a formal step-by-step approach to data capture, storage, and retrieval for use throughout the plant's life cycle a plant configuration management system (PLANT/CMS)

416
Recent inspection results in the electrical and instrumentation and control areas. Athavale, S V , Imbro, E V , Haughney, C J *Transactions of the American Nuclear Society (USA),* 60 450 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

The US Nuclear Regulatory Commission (NRC) inspection program includes a comprehensive examination of the design, installation, and testing aspects of selected modifications Conclusions about the overall modification control process, including the design, procurement, testing, start-up, maintenance, training, and operation may then be drawn based on the results of the reviewed samples The inspections are multidisciplinary reviews that include as a minimum, the mechanical systems, electrical power, and instrumentation and control (I and C) disciplines The paper presents examples of concerns identified by the NRC team during past inspections NRC s inspection program not only verifies the licensee s commitment to regulatory requirements but also analyzes, evaluates, and verifies, on an audit basis, that plant modifications have been designed, engineered, installed, and tested in such a manner that plant safety is not compromised

417
Fire protection in the reactor building, nuclear power station, decommissioning order. *ET, Energiewirtschaftliche Tagesfragen (Germany, F R),* 40 No 7, 508-510 (Jul 1990) (In German)

The action of Preussen Elektra against the immediate implementation of the temporary shutdown of the Wuergassen reactor (KWW) by the supervising authority because of insufficient fire protection in the reactor containment (SB) was given non-suit To render possible the continued operation of the reactor, the plaintiff had proposed to use as systems to fight fire the pressure chamber spray system of the pressure attenuation system (DAS) with a capacity of twice 600 t/h until inertisation of the reactor containment In the opinion of the court, use of the pressure chamber spray system for containing accidents upon loss of coolant, being a fire extinguishing system using water constitutes an essential modification of operation within the meaning of the Atomic Energy Act (AtG), articles 7, section 1, and requires a (modification) permit The reasons for the non-suit are presented in detail Higher Administrative Court of North-Rhine Westphalia, decree of January 2, 1990 - AZ 21 D 66/89 AtG articles 7, sections 1, and 19, subsection 3 (orig /HP)

418
Official notice concerning the publication of a licence issued under nuclear law, for modifications to be made in the Obrigheim reactor. As of September 26, 1990. (Ministerium fuer Wirtschaft, Mittelstand und Technologie des Landes Baden-Wuerttemberg, Stuttgart (Germany, F R)) *Bundesanzeiger (Germany, F R),* 42 No 187, 5267 (6 Oct 1990) (In German)

Published in summary form only OBRIGHEIM REACTOR/modifications, MODIFICATIONS, CONSTRUCTION PERMITS, LEGAL ASPECTS, LICENSING

419
Official announcement of place of publication of a license issued under nuclear law, for modifications to be made in the Philippsburg nuclear reactor station. As of October 8, 1990. (Ministerium fuer Wirtschaft, Mittelstand und Technologie des Landes Baden Wuerttemberg, Stuttgart (Germany, F R)) *Bundesanzeiger (Germany, F R),* 42 No 194, 5486 (17 Oct 1990) (In German)

Published in summary form only PHILIPPSBURG-1 REACTOR/ modifications, MODIFICATIONS, LICENSING, LEGAL ASPECTS

ECONOMICS

420
(GAO/RCED--89-88FS)
Nuclear science. (General Accounting Office, Washington, DC (USA) Resources, Community and Economic Development Div) Apr 1989 11p US General Accounting Office, P O Box 6015, Gaithersburg, MD 20877 (USA)

This report answers questions about the Department of Energy's possible acquisition and conversion of a partially completed commercial nuclear power plant to a nuclear materials production facility The nuclear power plant is the Washington Nuclear Plant No 1 owned by the Washington Public Power Supply System and is located on DOE's

Hanford Reservation near Richland Washington

421
Nuclear power development. Nealey S Columbus OH (USA) Battelle Press (1990) 82p Battelle Press 505 King Avenue Columbus OH 43201 (USA)

The objective of this study is to examine factors and prospects for a resumption in growth of nuclear power in the United States over the next decade The focus of analysis on the likelihood that current efforts in the United States to develop improved and safer nuclear power reactors will provide a sound technical basis for improved acceptance of nuclear power, and contribute to a social/political climate more conducive to a resumption of nuclear power growth The acceptability of nuclear power and advanced reactors to five social/political sectors in the U S is examined Three sectors highly relevant to the prospects for a restart of nuclear power plant construction are the financial sector involved in financing nuclear power plant construction the federal nuclear regulatory sector and the national political sector For this analysis the general public are divided into two groups those who are knowledgeable about and involved in nuclear power issues the involved public and the much larger body of the general public that is relatively uninvolved in the controversy over nuclear power

422
East German nuclear power projects face an uncertain future. Varley J *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432, 12, 14 15 (Jul 1990)

Units 1-4 at Greifswald (East Germany) are Soviet designed VVER 110 PWRS of the early V 230 type A decision to close them down was taken in the light of an assessment of plant safety carried out jointly by GRS the West German safety analysis organization, and SAAS the East German safety authority The assessment follows on from a preliminary' GRS report published in February this year which highlighted the problem of vessel embrittlement The new report focuses on operating experience and safety system design There are significant deficiencies in the plant s operating instructions and over the past decade there have been serious breaches of the instructions, including operation of the plant at full power with a leak in the primary circuit purification system outside the sealed zone of the reactor building, insufficient inspection, inadequate testing of modifications to important safety systems, misoperation of reactor safety systems, inadequacies in plant condition monitoring As regards the safety systems, some 75 measures are listed as being immediately necessary for continued operation, another 50 or so would allow operation for a limited period, while 20 more would be needed for long term operation They include improved emergency cooling and power, instrumentation upgrades, organizational improvements, and better fire protection (author)

423
Nuclear power fizzles out (the future of the nuclear industry in the UK). Wyman, V *Engineer (London) (UK),* 270 No 7002, 30 (7 Jun 1990)

The UK nuclear energy programme has effectively been put on hold until 1994 The question as to whether the loss of impetus has damaged the industry so badly that it cannot respond to the needs of the future is addressed (author)

424
Starting up a district heat reactor in China. Wang Dazhong *Nuclear Engineering International (Incorporates Nuclear Power) (UK)* 35 No 431, 15 (Jun 1990)

A 5MWt light water nuclear plant - the world s first dedicated nuclear district heating plant in regular operation - has been in use in China for six months The reactor start-up s described (author)

425
The benefits of openness at Cernavoda. Karger, J *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 431, 18-19 (Jun 1990)

Increasing openness and cooperation on the part of the new Romanian government has brought benefits all round in the Cernavoda project, where five CANDU nuclear reactors are under construction It is now thought that the first unit could be put into operation by 1993, with the final units being completed around 1999 (author)

426
JAIF compiles report on nuclear industry market trend in Japan. *Atoms in Japan (Japan),* 34 No 6, 24-26 (Jun 1990)

Published in summary form only NUCLEAR POWER PLANTS/japan, JAPANESE ORGANIZATIONS, NUCLEAR INDUSTRY, JAPAN, CONSTRUCTION

427
FY 1989 nuclear capacity factor at 70.0 %, incidents and failures: 0.6 per unit. *Atoms in Japan (Japan),* 34 No 4, 21-23 (Apr 1990)

Published in summary form only NUCLEAR POWER PLANTS/capacity, REACTOR OPERATION, JAPAN, REACTOR SHUTDOWN, BWR TYPE REACTORS, PWR TYPE REACTORS, CAPACITY, INSPECTION, ANNUAL VARIATIONS, FAILURES, STATISTICAL DATA

428
Nuclear power in the United States and around the world. McKlveen, J W , Barnett, J M , Delahunte, K E *Radiation Protection in Australia (Australia),* 8 No 1, 18-22 (Jan 1990) (CONF-890669–)

From 14 annual conference of the Australian Radiation Protection Society, Inc , Perth (Australia) (26-29 Jun 1989)

The current status and projected number of future reactors in the US and around the world are presented Since nuclear power plants have become increasingly costly and complicated to built, research has begun on advanced reactors The two advanced reactor concepts described are the light water and gas cooled designs It is stressed that these reactors are still in the design phase and that questions do remain about their safety 4 refs , 9 tabs

429
Cost of decommissioning nuclear power plants. De P L *International Atomic Energy Agency Bulletin (Austria),* 32 No 3, 39-42 (1990)

Over the past 35 years, considerable experience has been gained in decommissioning many types of nuclear facilities By the turn of the century, more than 60 nuclear power plants and 250 research reactors around the world will become likely candidates for decommissioning Several factors influence the choice of decommissioning strategy in a country and consequently, the decommissioning cost estimates performed are difficult to compare International efforts to harmonize the various estimates are essential The IAEA introduced the concept of various cost elements or components, a suggested methodology and it has also undertaken in its 1991-92 programme, some specific studies on a common methodology for decommissioning cost estimation 2 figs, 1 map, 2 tabs

430
Nuclear desalination: Experience, needs, and prospects. Barak, A , Kochetkov, L A , Crijns, M J , Khalid, M *International Atomic Energy Agency Bulletin (Austria),* 32 No 3, 43 48 (1990)

The article reviews the USSR s experience with nuclear desalination as well as a joint project of the United States and Israel, and recent studies in the United States, Japan, and the Federal Reoublic of Germany

431
Ties between the energy and ecological programmes. Erban P , Horacek, P *Zpravodaj VUPEK (Czechoslovakia),* No 3, 20-26 (1989) (In Czech)

Power generation trends are analyzed with respect to the energy demands of the gross domestic product (GDP) In economically successful countries, the share of electricity in the total energy input into GDP is increasing In 1984 the gross electricity consumption contributed 12 to 35% to the total energy consumption in developed countries,whereas in Czechoslovakia this share was a mere 8 76% while the power consumption (in kWh per created GDP $) was 216% with respect to the values attained in developed European countries The low share of electricity in the total energy consumption thus was due to a much too high consumption of all forms of energy, direct consumption of fuels in particular In 1980 and 1984, the share of nuclear power plants in electricity generation in Czechoslovakia was 6 22 and 14 61%, respectively Problems associated with industrial emissions, of carbon dioxide in particular, are outlined It is concluded that (i) the structure of Czechoslovak industrial production should be altered as soon as possible to achieve a reduction in the raw material and energy demands, (ii) for many years it will be electrification that should enable the effectivity of Czechoslovak economy to be increased and the total energy demand of industrial production to be reduced, and (iii) increase in the share of nonfossil electricity generation is desirable for environmental reasons, increase in the share of natural gas as a substitute for coal is desirable as well (P A) 3 figs , 3 tabs

432
Nuclear power and acceptation: Problems about the deadlock. Speelman, J E *Energiespectrum (Netherlands),* 14 No 4, 109-113 (Apr 1990) (CONF-890841-)

From International workshop on safety of nuclear installations of the next generation and beyond, Chicago, IL (USA) (28-31 Aug 1989)

In 1989 a workshop was held organized by the IAEA and the Argonne National Laboratory The purpose was to investigate under which circumstances a large-scale extension of nuclear power can be accepted Besides the important technical information the care for the environment determined the atmosphere during the workshop The opinion dominated that nuclear power can contribute in tackling the environment problems, but that the social and political climate this almost makes impossible (author) 7 refs , 1 fig , 1 tab

433
Main regulations of technical and economic analysis of NPPs with variable loads. Boldyrev, V M , Lozgachev, V P *Izvestiya Akademii Nauk SSSR, Ehnergetika i Transport (USSR),* No 6, 158-164 (Nov-Dec 1989) (In Russian)

The main regulations of technical and economic analysis of NPPs with variable loads are presented Particular attention is paid to the methods of bringing different NPPs with energy storage systems to the unified energy effect without using supplementary ogranic-fueled power units with variable loads Methods for transformation of the daily electron load schedule to the calculated one by the equivalenting method is described The types of NPPs with variable loads are considered

434
Application of the discounted value flows method in production cost calculations for Czechoslovak nuclear power plants. Majer, P *Jaderna Energie (Czechoslovakia),* 36 No 6, 217-223 (Jun 1990) (In Czech)

English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per typewritten page

The fundamentals are outlined of the discounted value flows method, which is used in industrial countries for calculating the specific electricity production costs Actual calculations were performed for the first two units of the Temelin nuclear power plant All costs associated with the construction, operation and decommissioning of this nuclear power plant were taken into account With a high degree of certainty, the specific production costs of the Temelin nuclear power plant will lie within the range of 0 32 to 0 36 CSK/kWh Nearly all results of the sensitivity analysis performed for the possible changes in the input values fall within this range An increase in the interest rate to above 8% is an exception, this, however, can be regarded as rather improbable on a long-term basis Sensitivity analysis gave evidence that the results of the electricity production cost calculations for the Temelin nuclear power plant can be considered sufficiently stable (Z M) 7 figs , 2 tabs , 14 refs

435
Does nuclear power have a future?. Sweet, Colin *Energy Policy (UK),* 18 No 5, 406-422 (Jun 1990)

The withdrawal of the nuclear sector from the privatization process on 9 November 1989 threw a question mark over the future of nuclear power in the UK It also provided insights into the relationship between government and the nuclear industry This article explains the constructs by which the published costs of nuclear power were made to diverge sharply from the real costs It also analyses present and future costs, summarizing evidence given at the Hinkley Point C Public Inquiry The issue for decision makers, it is suggested, is one of damage limitation (author)

436
Stretch rating, a means to maintaining nuclear power competitiveness. Malek, J D , Nicholson, J M , Banerjee, A K *Transactions of the American Nuclear Society (USA),* 60 466 (1989) (CONF-891103-)

From Winter meeting of the American Nuclear Society (ANS) and nuclear power and technology exhibit, San Francisco, CA (USA) (26-30 Nov 1989)

In the midst of the changing economics of power generation, electric utilities are busily preparing and positioning themselves to be the low-cost producer of electricity This activity reflects the expectation of a deregulated electric power industry One means of increasing plant output without investing capital into new facilities is through the extended operation of nuclear power plants into their stretch rating Stone and Webster has performed several evaluation to determine the feasibility of uprating reactor core power output to the unit's stretch power rating This paper provides a summary of experiences and approaches to core power uprating/stretch rating of nuclear power plants The approach to performing core uprate analyses is multiphased which involves nuclear steam supply system evaluations and balance of-plant assessments The phases through which a core uprate progresses are the following (1) feasibility study, (2) detailed engineering evaluations (3) engineered plant modifications, and (4) core uprate implementation These phases represent a logical progression of the evaluation process from the initial economic cost/benefit analyses through the implementation of specific plant modifications, if required, to the licensing submittals and actual power ascension

FUEL CYCLE

437
(DOE/EIA-0436(90))
World nuclear fuel cycle requirements 1990. (USDOE Energy Information Administration Washington DC (USA) Office of Coal, Nuclear Electric and Alternate Fuels) 26 Oct 1990 77p NTIS, PC A05/MF A01 - GPO, OSTI, INIS, GPO Dep Order Number DE91001973

This analysis report presents the projected requirements for uranium concentrate and uranium enrichment services to fuel the nuclear power plants expected to be operating under three nuclear supply scenarios Two of these scenarios, the Lower Reference and Upper Reference cases, apply to the United States, Canada, Europe, the Far East, and other countries with free market economies (FME countries) A No New Orders scenario is presented only for the United States These nuclear supply scenarios are described in Commercial Nuclear Power 1990 Prospects for the United States and the World (DOE/EIA-0438(90)) This report contains an analysis of the sensitivities of the nuclear fuel cycle projections to different levels and types of projected nuclear capacity, different enrichment tails assays, higher and lower capacity factors, changes in nuclear fuel burnup levels, and other exogenous assumptions The projections for the United States generally extend through the year 2020, and the FME projections, which include the United States, are provided through 2010 The report also presents annual projections of spent nuclear fuel discharges and inventories of spent fuel Appendix D includes domestic spent fuel projections through the year 2030 for the Lower and Upper Reference cases and through 2040, the last year in which spent fuel is discharged, for the No New Orders case These disaggregated projections are provided at the request of the Department of Energy's Office of Civilian Radioactive Waste Management

MISCELLANEOUS

438
(CONF 9010235--1)
Use of neural networks in the operation of nuclear power plants. Uhrig, R E (Oak Ridge National Lab , TN (USA)) [1990] Contract AC05-84OR21400 ,FG07-88ER12824 7p NTIS, PC A02/MF A01 - OSTI, GPO Dep Order Number DE91002326

From American Association of Artificial Intelligence conference, Dayton, OH (USA) (29 31 Oct 1990)

Application of neural networks to the operation of nuclear power plants is being investigated under a US Department of Energy sponsored program at the University of Tennessee Projects include the feasibility of using neural networks for the following tasks (a) diagnosing specific abnormal conditions, (b) detection of the change of mode of operation, (c) signal validation, (d) monitoring of check valves, (e) modeling of the plant thermodynamics, (f) emulation of core reload calculations, (g) analysis of temporal sequences in NRC's "licensee event report," (h) monitoring of plant parameters, and (i) analysis of plant vibrations Each of these projects and its status are described briefly in this article the objective of each of these projects is to enhance the safety and performance of nuclear plants through the use of neural networks 6 refs

439
(EPRI-MD-6881 Vol 2, pp 15 1 15 17)
Methods for testing the logical structure of plant procedure documents. Horne, C P , Colley, R , Fahley, J M (Charles P Horne, Inc , Orinda, CA (USA)) Jul 1990 422p Research Reports Center, Box 50490, Palo Alto, CA 94303 (CONF-8912108-Vol 2)

From Conference on advanced computer technology for the power industry, Scottsdale, AZ (USA) (4-6 Dec 1989)

In Proceedings 1989 conference on advanced computer technology for the power industry Volume 2, Computer technologies

This paper describes an ongoing EPRI project to investigate computer based methods to improve the development, maintenance, and verification of plant operating procedures This project began as an evaluation of the applicability of structured software analysis methods to operating procedures It was found that these methods offer benefits, if procedures are transformed to a structured representation to make them amenable to computer analysis The next task was to investigate methods to transform procedures into a structured representation The use of natural language techniques to read and compile the procedure documents appears to be viable for this purpose and supports conformity to guidelines The final task was to consider possibilities of automated verification methods for procedures Methods to help verify procedures were defined and information requirements specified These methods take the structured representation of procedures as input The software system being constructed in this project is called PASS, standing for Procedures Analysis Software System

440
(EPRI-MD-6881-Vol 2, pp 16 1-16 11)
Using the polynomial discriminant method to classify nuclear power plant data. Alguindigue, I E , Uhrig, R E (Univ of Tennessee, Knoxville (USA)) Jul 1990 422p Research Reports Center, Box 50490, Palo Alto, CA 94303 (CONF-8912108-Vol 2)

From Conference on advanced computer technology for the power industry, Scottsdale, AZ (USA) (4-6 Dec 1989)

In Proceedings 1989 conference on advanced computer technology for the power industry Volume 2 Computer technologies

The Polynomial Discriminant Method PDM was used to classify power data from a pressurized water reactor nu clear plant The data consisted of reactor coolant system water flow read ings collected in each of the four loops of the plant The network was trained to discriminate between data collected during normal steady-state operating conditions and data collected during transient conditions during shut down The network was able to classify cor rectly over 97% of the cases in which it was tested Of particular importance was the network s ability to detect devi-ations from normal settings at the early stages of the transient The algorithm although computational intensive dur ing training exhibits great extrapolating abilities This procedure is amenable to on line operation in a nuclear power plant as a means of detecting the be ginning of transients

441

(IAEA-TECDOC 561)

Reviewing computer capabilities in nuclear power plants. (International Atomic Energy Agency Vienna (Aus-tria)) Jun 1990 87p NTIS (US Sales Only) PC A05/MF A01 OSTI, INIS Or-der Number DE91610110

The OSART programme of the IAEA has become an effective vehicle for promoting international co operation for the enhancement of plant opera tional safety In order to maintain consistency in the OSART reviews OS ART Guidelines have been developed which are intended to ensure that the reviewing process is comprehensive Computer technology is an area in which rapid development is taking place and new applications may be computerized to further enhance safety and the effectiveness of the plant Sup plementary guidance and reference material is needed to help attain com prehensiveness and consistency in OSART reviews This document is de-voted to the utilization of on-site and off site computers in such a way that the safe operation of the plant is sup-ported In addition to the main text, there are several annexes illustrating adequate practices as found at various operating nuclear power plants Refs, figs and tabs

442

(IAEA TECDOC–570)

OSART mission highlights 1988-1989. (International Atomic Energy Agency Vienna (Austria)) Sep 1990 111p NTIS (US Sales Only) PC A06/MF A01 OSTI INIS Order Num ber DE91610111

The IAEA Operational Safety Review Team (OSART) programme provides advice and assistance to Member States for enhancing the operational safety of nuclear power plants The ob servations of the OSART members are documented in technical notes which are then used as source material for the official OSART Report submitted to the government of the host country The technical notes contain recom mendations for improvements and description of commendable good practices The same notes have been used to compile this summary report This report is the third in a series following IAEA TECDOC 458 and IAEA TECDOC-497 and covers the period June 1988 to May 1989

443

Plex approach to nuclear power plant aging evaluation and compo-nent life prediction. Kaushansky, M M pp 1049-1062 of Power-gen 1989 Conference papers, Volumes V and VI Houston, TX (US), Power-Gen (1989) 413p (CONF-891217–)

From POWER GEN '89 2nd confer-ence and exhibition for the power generation industries, New Orleans, LA (USA) (5 7 Dec 1989)

Plant Life Enhancement/Extension (PLEX) in the Westinghouse view, is a planned program of ongoing modern-izations and improvements that can extend the period of peak plant avail-ability and the total service life of the plant over time The prediction of major plant components life span establish-ment of their mean time to failure (MTTF) and equipment replacement or modification planning procurement and installation (before failure) is the main goal of a Westinghouse PLEX program

444

Operational improvements through design simplification and design margin evaluations. Asztalos, M J pp 1115-1140 of Power-gen 1989 Conference papers, Volumes V and VI Houston, TX (US), Power-Gen (1989) 413p (CONF-891217–)

From POWER GEN '89 2nd confer-ence and exhibition for the power generation industries New Orleans LA (USA) (5 7 Dec 1989)

Systems design simplification is the process of modifying the design (e g the hardware, the operation the qualifi cation requirements, the operational paperwork) to simplify the operation of the nuclear power plant Design simpli fication has been an evolutionary process that has become feasible due the accumulation of operating experi ence diversity of equipment designs and vendors improved industry codes standards and analytical tools Exam ples of design simplification that have been successfully implemented include the use of drag valves for high fluid pressure drop applications removal of concentrated sodium hydroxide from containment pressure suppression sys tems or retiring in place subsystems that were designed for redundancy but have proved to be unnecessary Sys tems margin evaluations are processes used to evaluate a systems (or its com ponents) performance against the systems functional requirements and identifying the performance margins These margins are available due to the very conservative design assumptions used to design nuclear power systems, design and manufacture nuclear com-ponents, and erect the nuclear power plant Examples of areas of margin re duction include electronic systems response times, concentration reduc-tions of reactivity control solutions, and reductions of performance require-ments for engineered safeguards pumping capacities This paper will evaluate the major systems design simplification areas and available sys tems margins that have successfully been utilized to improve plant availabil-ity and operation

445

Strategy and planning for decom-missioning of UKAEA facilities. Flowers R H , Williams, J *Atom (London) (UK),* No 405 6-8 (Jul-Aug 1990) (CONF-900387–)

From 2 international seminar on de commissioning of nuclear facilities London (UK) (19-20 Mar 1990)

There are many important factors to consider when defining the priority of a decommissioning task These include safety, the cost of dismantling versus maintenance, waste management costs, re-use of the facility, depen-dence on other facilities, licensing and the availability of experienced staff In this paper, the evaluation process used to categorise the phasing of the decommissioning of UK Atomic Energy Authority facilities is described (author)

446
Decommissioning enters the commercial era. Kovan, D *Atom (London) (UK),* No 405 9-12 (Jul-Aug 1990) (CONF-900387–)
From 2 international seminar on decommissioning of nuclear facilities, London (UK) (19-20 Mar 1990)
Judging by some stories that have appeared in the press the public could be excused for thinking that it was decommissioning that ultimately forced the government to retain nuclear power in the public sector The development of this confusion formed the backdrop to the discussions reported here (author)

447
Analysing load factors by vendor. Knox R *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 432 20 22 (Jul 1990)
A further analysis of the 1989 figures for nuclear plant load factors published previously is presented Performance trends by vendor are shown (author)

448
New emphasis on safety. Masters, R *Nuclear Engineering International (Incorporates Nuclear Power) (UK),* 35 No 431, 20-21, 23-28 (Jun 1990)
Nuclear power continues to make a major contribution to world electricity supplies But the capacity of new reactors entering service and new construction starts in the past twelve months was almost exactly offset by the capacity of reactors shut down for decommissioning and halts on construction The details are given in a country by-country survey of developments in 1989 The most notable development in the nuclear scene in the past year has been the reassessment of the safety of the Soviet reactors in operation and under construction in Eastern European countries following the political changes there This is a major setback to their nuclear plans even though it opens up prospects for work from Western companies for safety systems and complete reactors The general commitment to improving performance and safety standards was clearly emphasised in the signing of the charter of the World Association of Nuclear Operators (WANO) by 138 utility executives, representing between them every operating nuclear plant in the world (author)

449
Dismantling of nuclear power plants. Boekschoten, H J C *Energiespectrum (Netherlands),* 14 No 3, 83-87 (Mar 1990) (In Dutch)
One of the most important discussion points with regard to wether or not extending the Dutch nuclear-power programme is the possibility for dismantling of nuclear power plants at the end of their operation time Full dismantling of a large nuclear power plant as will have to be done also in Western-Europe, has thus far not been demonstrated However, technology is progressing so much that no large technical difficulties are expected In some countries dismantling is delayed because there does not exist a final destination for the radioactive dismantling wastes (author) 7 refs , 6 figs , 1 tab

450
Assessment of nuclear power plant site selection by comparative method. Grosikova, B , Uvirova, E , Stranai, I *Radioaktivita a Zivotne Prostredie (USSR),* 12 No 5, 227-230 (1989) (In Slovak)
English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per page
An optimization method of selecting alternative sites for nuclear power plants planned in Czechoslovakia, and a detailed algorithm of its application are described (author) 5 refs

451
Small nuclear heating reactors - cost-effectiveness and market potential in Switzerland. Klockow, S *District Heating International (Germany, F R),* 19 No 5, 369-377 (Sep-Oct 1990) (In German)
Nuclear heating reactors (NHR) are designed for the production of heat with a relatively low temperature (120deg C) and under low pressure As these reactors can be built near the heat consumer, the distribution costs of district heat generated in heating reactors are comparatively low Although most concepts for nuclear heating reactors data from the seventies, only a few prototypes are under operation Against the background of increasing environmental problems, especially the greenhouse-effect, the generation of district heat in nuclear heating reactors could gain a greater significance in the future In a recently conducted study the cost effectiveness and the market potential for district heat from heating reactors in Switzerland were examined Because the heat demand of the industrial sector requires detailed analysis of each individual plant, the study was confined to the residential and service sector In accordance with Switzerland's settlement structure, the study focused on two types of small heating reactors (GHR and Geyser) with a capacity of 10 to 50 MW (orig)

452
Standardization for power stations in the German Democratic Republic (GDR). Roesler, J *VGB Kraftwerkstechnik (Germany, F R),* 70 No 9, 742-748 (Sep 1990) (In German)
This paper describes standardization in the field of power plant engineering in the GDR from the point of view of a practical man with 20 year's experience as an employee of a central state organization for standardization There is no point in comparing the system of technical regulations in the GDR with western standards - that is done only in cases in which it is indispensable for a proper understanding (orig)