AVALIAÇÃO EXPERIMENTAL DA CONVEÇÃO NATURAL E DA ALTURA DO EFEITO CHAMINÉ NO REATOR NUCLEAR DE PESQUISA TRIGA IPR-R1

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Resumo: Os reatores nucleares de pesquisa tipo TRIGA (Training, Research, Isotopes, General Atomic) são considerados inerentemente seguros devido, principalmente, ao seu sistema passivo de remoção do calor gerado pelas combustíveis. O Reator TRIGA IPR-R1 do Centro de Desenvolvimento da Tecnologia Nuclear (Belo Horizonte) opera a uma potência térmica estacionária máxima de 250 kW. O núcleo do reator situa-se em um poço cilíndrico de cerca de 6 m de profundidade. A refrigeração dos elementos combustíveis no núcleo é feita pela circulação natural de água leve ao longo dos canais de refrigeração. Aqui são descritos os experimentos realizados com a finalidade de conhecer o mecanismo de convecção natural no núcleo e ao longo da piscina. Para a medida de temperatura nos canais de refrigeração do núcleo, utilizaram-se duas sondas com termopares. Para a medida de temperatura no poço, nove termopares foram distribuídos axialmente ao longo do poço. Os sinais foram monitorados, em tempo real, na tela do computador do sistema de aquisição de dados. Os resultados obtidos foram comparados tanto com experimentos semelhantes, realizados em outros reatores, quanto com previsões teóricas. A altura do efeito chaminé é considerada como a distância entre a saída do canal e o plano das isothermas do fluido acima do núcleo. A altura do efeito chaminé foi avaliada experimentalmente em função dos níveis de potências de operação do reator.

Palavras-chave: TRIGA, reator nuclear de pesquisa, convecção natural, combustível nuclear.

1. INTRODUÇÃO

O Reator Nuclear TRIGA IPR-R1 (Fig. 1), do Centro de Desenvolvimento da Tecnologia Nuclear – CDTN, é um modelo Mark I, fabricado pela General Atomic de San Diego – Califórnia, refrigerado por água leve desmineralizada e tendo como combustível urânio enriquecido a 20% em 235U. Foi projetado para treinamento, pesquisa, ativação neutrônica de materiais e produção de radioisótopos. Os reatores TRIGA são caracterizados pela sua segurança intrínseca devido, principalmente, ao grande coeficiente negativo de temperatura/reactividade. Isso significa que um aumento da potência leva a um consequente aumento da temperatura da mistura combustível-moderador, causando o aparecimento de uma reatividade negativa que amortece gradualmente a taxa de aumento de potência e esta tende a se estabilizar. Outras características de segurança dos reatores TRIGA são a alta retenção dos produtos de fissão no combustível, mesmo que o revestimento venha a sofrer falha, e um sistema passivo de remoção de calor no núcleo durante as operações. O IPR-R1 é um reator nuclear de pesquisa do tipo piscina, refrigerado por circulação natural. O calor acumulado na água do poço pode, opcionalmente, ser removido por circulação forçada por um sistema de refrigeração dotado de circuito primário e circuito secundário.

As principais formas de utilização do Reator TRIGA IPR-R1 são a ativação neutônica e a produção de radioisótopos sendo, pois, otimizado para a utilização dos fluxos de neutrons. Várias pesquisas na área de neutrônica já foram realizadas em suas instalações, mas existem carências de dados, principalmente experimentais, sobre seu comportamento termo-hidráulico. Realizaram-se pois, medidas de temperaturas nos canais de refrigeração do núcleo e ao longo da piscina, com o reator operando em vários níveis de potência. Assim, puderam-se avaliar experimentalmente vários parâmetros da refrigeração natural, como a vazão do refrigerante no núcleo e a altura do efeito chaminé, em função das potências de operação. A altura do efeito chaminé, neste caso uma chaminé virtual, pois ela não existe fisicamente, é considerada como a distância entre a saída do canal e o plano das isothermas do fluido acima do núcleo.
2. MEDIDA DE TEMPERATURA NO NÚCLEO

A monitoração de temperatura no núcleo do reator foi feita por meio de dois termopares, posicionados em dois canais distintos, sendo um à entrada (TMP 7) e o outro à saída do canal (TMP 6), conforme mostrado na Fig. 2. Os dois termopares foram fixados em duas sondas de alumínio, que foram apoiadas na placa superior do núcleo do reator e desceram através de orifícios de 8 mm de diâmetro existentes nesta placa. As sondas possuem diâmetros de 7,9 mm, para não permitir sua inclinação dentro do canal de modo a não tocarem as paredes dos combustíveis. A primeira sonda foi construída com um comprimento tal que permite posicionar o termopar logo abaixo do início do comprimento aquecido, atravessando verticalmente toda a extensão do canal. Para o levantamento do perfil de temperatura ao longo do canal, esta sonda foi erguida em passos de 5 cm em 5 cm. A segunda sonda, apresenta um comprimento suficiente apenas para posicionar o termopar imediatamente acima do comprimento aquecido. O objetivo deste posicionamento é permitir a obtenção da variação da temperatura da água ao percorrer o núcleo do reator.

Figura 2. Localização dos termopares no núcleo do reator.
3 MEDIDA DE TEMPERATURA NO POÇO

A monitoração de temperatura no poço do reator foi realizada por meio de nove termopares e um termoresistor. Nove termopares foram fixados em uma sonda vertical de alumínio, sendo que o primeiro deles (TMP 7) ficou a 143 mm acima da placa superior do núcleo, os outros estão fixados acima deste (Fig. 3). Para o valor de temperatura abaixo do núcleo considerou-se a medida do termoresistor (AI 2), que mede a temperatura de entrada no circuito primário de refrigeração forçada. O termoresistor (AI 21) se encontra fixado a 2 m acima do núcleo, e é o sensor que normalmente fornece a temperatura do poço para a mesa de controle.

Um termopar (TMP 4) ficou a cerca de 30 cm acima do nível do poço para medida da temperatura ambiente. Dois termopares (TMP 5 e TMP 18) ficaram a 3 m de profundidade nos poços situados ao lado do tanque do reator, para medida da temperatura do solo. Dois termoresistores (AI 2 e AI 3) estão localizados, respectivamente, na entrada e saída do sistema de refrigeração forçada do poço.

Figura 3. Localização dos sensores de temperatura no poço do reator.

4 RESULTADOS

4.1 Fluxo de Massa de Refrigerante no Canal Quente do Núcleo

Na Figura 4 tem-se em destaque o Canal 1', que teve sua temperatura monitorada e o Canal 0 situado mais no centro onde se tem a maior densidade de fluxo neutrônico. Não existe orifício na placa superior do núcleo na direção do Canal 0, portanto não foi possível medir sua temperatura. As temperaturas de entrada e saída no Canal 0 serão consideradas como sendo iguais às do Canal 1'. Os dados geométricos do Canal 0 e do Canal 1 são dados na Tabela 1.

O aquecimento de cada canal é resultado da soma das contribuições das potências das frações dos perímetros dos combustíveis em torno do canal. A potência total do núcleo é de 265 kW, dando uma média de 4,518 kW para cada elemento combustível com revestimento de aço e 4,176 kW para cada elemento com revestimento de alumínio. Os valores são multiplicados pelos fatores radiais de distribuição de potência no núcleo dado pelo cálculo neutrônico e multiplicados pelo fator de distribuição axial de potência no combustível (1,25). Os produtos são multiplicados pelas frações dos perímetros de cada combustível em contato com o canal.
Figura 4. Canais mais aquecidos do núcleo do reator.

Na Tabela 1 tem-se os dados do Canal 0 e do Canal 1′ e o percentual de contribuição de potência de cada combustível para o aumento de temperatura da água ao longo deles.

<table>
<thead>
<tr>
<th></th>
<th>Canal 0</th>
<th>Canal 1′</th>
<th>Unidade</th>
</tr>
</thead>
<tbody>
<tr>
<td>Área (A)</td>
<td>1,574</td>
<td>8,214</td>
<td>cm²</td>
</tr>
<tr>
<td>Perímetro Molhado (Pw)</td>
<td>5,901</td>
<td>17,643</td>
<td>cm</td>
</tr>
<tr>
<td>Perímetro Aquecido (Ph)</td>
<td>3,906</td>
<td>15,156</td>
<td>cm</td>
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<tr>
<td>Diâmetro Hidráulico (Dw)</td>
<td>1,067</td>
<td>1,862</td>
<td>cm</td>
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<tr>
<td>Diâmetro do Combustível B1 e C11 (inox.)</td>
<td>3,76</td>
<td>3,76</td>
<td>cm</td>
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<tr>
<td>Diâmetro do Combustível B6 e C12 (Al)</td>
<td>3,75</td>
<td>3,75</td>
<td>cm</td>
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<tr>
<td>Diâmetro da Barra de Controle C1</td>
<td>3,80</td>
<td>3,80</td>
<td>cm</td>
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<tr>
<td>Diâmetro do Tubo Central</td>
<td>3,81</td>
<td>3,81</td>
<td>cm</td>
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<tr>
<td>Potência Total do Núcleo (265 kW)</td>
<td>100</td>
<td>100</td>
<td>%</td>
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<tr>
<td>Contribuição do Combustível B1 (aço)</td>
<td>0,54</td>
<td>1,11</td>
<td>%</td>
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<tr>
<td>Contribuição do Combustível B6 (Al)</td>
<td>0,46</td>
<td>0,94</td>
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<tr>
<td>Contribuição do Combustível C11 (aço)</td>
<td>-</td>
<td>0,57</td>
<td>%</td>
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<tr>
<td>Contribuição do Combustível C12 (Al)</td>
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<td>1,08</td>
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<td>Total Potência no Canal</td>
<td>1,00</td>
<td>3,70</td>
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No processo de convecção natural que ocorre neste reator, as forças de circulação provêm das diferenças de densidades entre as camadas de água ao longo dos canais de refrigeração. Contra estas forças agem as perdas por expansão e contração das áreas de escoamento na entrada e saída do canal, as perdas de energia cinética e potencial e as perdas por atrito.

A água entra no canal pelos orifícios da placa inferior, percorre uma região não aquecida de grafite, passa pela região ativa retirando o calor do elemento combustível, passa na região de grafita não aquecida e sai do canal nos espaços existentes entre as cabeças dos elementos combustíveis e a placa superior.

Medidas diretas do fluxo de massa nos canais do núcleo não são possíveis por causa do pequeno tamanho dos canais, da baixa precisão dos medidores, além de ocorrer distúrbio na velocidade do fluido. O fluxo de massa no canal é dado pela vazão de massa dividida pela área do canal. A vazão de massa pode ser determinada indiretamente através do balanço térmico do fluido através do núcleo utilizando a medida da temperatura da água na entrada e na saída do canal, ou seja:

\[ q = \dot{m}c_p\Delta T \]

onde: \( q \) é a potência fornecida ao canal em [kW], \( \dot{m} \) é a vazão mássica no canal em [kg/s], \( c_p \) é o calor específico isobárico da água em [J/kgK] e \( \Delta T \) é a diferença de temperatura entre a entrada e a saída do canal em [°C].
Na Tabela 2 tem-se os dados do refrigente em função da potência fornecida aos Canais 1° e 0. Na tabela o fluxo de massa \( G \) é dado por: \( G = \dot{m} \cdot \text{área do canal} \). A velocidade \( u \) é dada por: \( u = G \rho \), sendo \( \rho \) a densidade da água (995 kg/m³). Os valores das propriedades termodinâmicas da água à pressão de 1,5 bar em função da temperatura média do fluido no canal foram estimados, por interpolação, da tabela fornecida por Wagner e Kruse (1998).

**Tabela 2. Propriedades do refrigente nos canais quentes do núcleo do reator.**

<table>
<thead>
<tr>
<th>q [kW]</th>
<th>q [kW]</th>
<th>( \Delta T ) [°C]</th>
<th>( c_p ) [kJ/kgK]</th>
<th>( \dot{m} ) [kg/s]</th>
<th>( G ) [kg/m²s]</th>
<th>( u ) [m/s]</th>
<th>( \mu ) [10⁻³kg/ms]</th>
<th>( k ) [W/mK]</th>
<th>Re</th>
<th>Pr</th>
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<td>Canal 1°</td>
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<td>Canal 0</td>
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**4.2 Perfil axial de temperatura no canal do núcleo**

Com as temperaturas do reator em equilíbrio térmico com o meio, operando nas potências de 106 kW e depois em 265 kW, a sonda do Canal 1°, que mede a temperatura da entrada deste canal, foi erguida da posição inferior até a saída do núcleo e a temperatura foi registrada. Os valores obtidos são mostrados no gráfico da Figura 5. No gráfico tem-se também resultados experimentais obtidos por Büke e Yavuz (2000), Bärs e Vaurio (1966) e Haag (1971) e os valores encontrados com código PANTERA.

Apesar do Canal 1° situar-se ao lado da barra de controle, o perfil axial de temperatura não sofreu influência de uma possível deformação do fluxo neutrônico provocado pela barra, pois esta se encontrava na sua posição superior, isto é, fora do núcleo.

**Figura 5. Variação da temperatura ao longo do Canal 1°.**

Na potência de 106 kW a temperatura atinge um valor máximo próximo à altura do centro do combustível. Em 265 kW este valor máximo ocorre um pouco acima do centro. Haag (1971) obteve uma curva similar, operando à potência de 1 MW. Uma possível explicação para este decréscimo da temperatura do fluido após passar pela parteativa do
combustível, teria a existência de fluxo cruzado de água vindo de canais mais frios na periferia do núcleo. Em todas as referências consultadas, para reatores de potência, o perfil ao longo do canal é sempre mostrado de modo crescente.

4.3 Distribuição de Temperaturas no Poço

O reator ficou crítico em 265 kW durante cerca de 8 h, quando então as temperaturas do poço entraram em equilíbrio térmico com o meio ambiente, isto é, observou-se que as várias temperaturas do poço mantiveram uma diferença constante em relação à temperatura da água e do lençol freático (o tanque do reator situa-se abaixo do nível do solo).

A Figura 6 mostra a evolução das temperaturas no poço, de entrada e saída do canal de refrigeração do núcleo mais próximos da posição do combustível B1 (Canal 1), até alcançarem o equilíbrio térmico com o meio ambiente. O estado estacionário só foi atingindo cerca de 7 horas após o início da operação. A resposta do termopar TMP 8 (Poço Sup 1), que mediria a temperatura na parte superior do poço, não está representada nos gráficos. Após os testes descobriu-se que seus termoelementos estavam em “curto-circuito” em algum ponto fora do poço, estando ele indicando a temperatura ambiente. Suas leituras foram descartadas.

O gráfico da Figura 7 mostra as temperaturas do poço e do canal de refrigeração do núcleo durante o equilíbrio térmico. Pode-se observar pelo gráfico que a temperatura do termopar situado a 143 mm acima da placa superior (Inf 7) registra uma temperatura em um patamar mais elevado do que os outros termopares localizados acima do núcleo, mostrando que o “efeito chaminé” tem uma altura muito pequena, no máximo 400 mm acima do núcleo, concordando com o trabalho experimental de Rao et al. (1988). Na Tabela 3 têm-se os valores médios medidos durante o regime permanente, isto é, durante a última 1 h do final da operação de 8 h em 265 kW.

Tabela 3. Temperaturas médias no poço e núcleo no equilíbrio térmico em 265 kW.

<table>
<thead>
<tr>
<th>Distância do Núcleo [mm]</th>
<th>Localização no Poço</th>
<th>Número do Sensor</th>
<th>Temperatura [°C]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Saída Primário</td>
<td>AI 3</td>
<td>35,1</td>
</tr>
<tr>
<td>5383</td>
<td>Ar</td>
<td>TMP 4</td>
<td>29,9</td>
</tr>
<tr>
<td>3083</td>
<td>Sup 2</td>
<td>TMP 9</td>
<td>44,6</td>
</tr>
<tr>
<td>2113</td>
<td>Inf 1</td>
<td>TMP 10</td>
<td>45,3</td>
</tr>
<tr>
<td>1783</td>
<td>Inf 2</td>
<td>TMP 11</td>
<td>46,6</td>
</tr>
<tr>
<td>1453</td>
<td>Inf 3</td>
<td>TMP 12</td>
<td>46,8</td>
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<tr>
<td>1123</td>
<td>Inf 4</td>
<td>TMP 13</td>
<td>47,4</td>
</tr>
<tr>
<td>793</td>
<td>Inf 5</td>
<td>TMP 14</td>
<td>48,0</td>
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<tr>
<td>463</td>
<td>Inf 6</td>
<td>TMP 15</td>
<td>48,6</td>
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<tr>
<td>143</td>
<td>Inf 7</td>
<td>TMP 16</td>
<td>57,1</td>
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<tr>
<td></td>
<td>Saída Canal 1</td>
<td>TMP 7</td>
<td>60,7</td>
</tr>
<tr>
<td></td>
<td>Entrada Canal 1’</td>
<td>TMP 6</td>
<td>47,1</td>
</tr>
<tr>
<td>-300 (abaixo do núcleo)</td>
<td>Entr Primário</td>
<td>AI 2</td>
<td>41,7</td>
</tr>
</tbody>
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Figura 6. Evolução das temperaturas no poço em 265 kW.
Figura 7. Temperaturas no poço durante o equilíbrio térmico (265 kW).

5 CONCLUSÕES

As medidas de temperatura no canal mais quente do núcleo mostraram que a temperatura da água atinge um máximo pouco acima do centro da parte ativa do elemento combustível, ocorrendo em seguida um pequeno decréscimo na temperatura. As medidas no poço mostraram que as temperaturas se tornam praticamente uniformes a uns poucos centímetros acima do núcleo, indicando que o efeito chamado “virtual”, definido como a altitude acima do núcleo na qual ocorre a circulação do fluido devido à diferença de densidade causada pela variação de temperatura, possui uma altura muito pequena e depende da potência de operação e da temperatura média do fluido.

Como se pode observar nas Figuras 6 e 7, o ponto onde se coleta a água para o circuito de refrigeração, é o local onde se mediu a temperatura mais baixa no poço, isto é, a captação de água deveria ser feita em um local acima do núcleo onde a água é mais quente, melhorando o rendimento. De qualquer modo as experiências mostraram a eficiência da circulação natural na remoção do calor gerado pelas fissões nucleares no núcleo. A circulação forçada tem a dupla finalidade de diminuir os níveis de radiação na sala do reator e diminuir a temperatura do poço quando se atinge o equilíbrio térmico com o meio.

6 AGRADECIMENTOS

Os autores expressam aqui seu agradecimento aos operadores do Reator Nuclear de Pesquisa TRIGA IPR-R1 pela ajuda durante os experimentos, a Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) e ao Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq) pelo apoio financeiro parcial.

7 REFERÊNCIAS


8 DIREITOS AUTORAIS

Os autores são os únicos responsáveis pelo conteúdo do material impresso incluído no seu trabalho.
EXPERIMENTAL STUDY OF NATURAL CONVECTION AND THE CHIMNEY EFFECT IN THE IPR-R1 NUCLEAR RESEARCH REACTOR

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Abstract. The TRIGA (Training, Research, Isotopes, General Atomic) nuclear reactors are considered inherently safe mainly due to its passive heat removal system. The IPR-R1 TRIGA Reactor of the Nuclear Technology Development Center (Belo Horizonte, Brazil) operates at a maximum thermal power of 250 kW. The reactor core is located at the bottom of a 6 m depth cylindrical tank. The fuel elements are cooled by light water natural circulation along the flow channels of the core. An experimental study has been carried out to monitor the natural convection mechanism in the Reactor. Two probes were used to measure the temperature in the core channels each one with a thermocouple. Nine thermocouples were distributed vertically along the reactor pool for water temperature measurements. These temperature measurements were show in real time in the computer screen of the data acquisition system. The experimental results were compared with others similar experiments and with theoretical analyses. It was also studied the chimney effect that is considered as an unheated extension of the core. The chimney height is the distance between the channel exit and the fluid isotherm plan above the core. This chimney height was evaluated as function of the reactor power

Keywords: TRIGA, research nuclear reactor, natural convection, nuclear fuel.
RESUMO

O Reator Nuclear de Pesquisa TRIGA IPR-R1 do Centro de Desenvolvimento da Tecnologia Nuclear, está localizado em Belo Horizonte (Brasil). Como todo reator TRIGA, seu núcleo está posicionado em uma piscina e os elementos combustíveis são refrigerados por circulação natural de água leve. A remoção de calor por este processo é eficiente na atual potência de 250 kW. Entretanto, existe um sistema forçado de remoção de calor da água do poço utilizado, principalmente, para reduzir o nível de radiação na sala do reator. A água é bombeada através de um trocador de calor, onde o calor é transferido do circuito primário para o circuito secundário. A monitoração de potência dos reatores nucleares é sempre realizada por instrumentos que medem o fluxo de nêutrons. No Reator IPR-R1 a potência é medida por quatro canais nucleares, cujos sensores primários são três câmaras de ionização e uma câmara de fissão. Este trabalho apresenta os resultados e a metodologia desenvolvida para monitoração, em tempo real, da potência do reator IPR-R1, por meios térmicos. Um dos métodos é a monitoração das temperaturas do elemento combustível e do poço. Outros dois métodos consistem no balanço térmico, em estado estacionário, dos circuitos primário e secundário de refrigeração forçada do reator.

PALAVRAS CHAVE: Potência, combustível nuclear instrumentado, reator nuclear TRIGA, termopares.

CÓDIGO: 61
INTRODUÇÃO

O Reator Nuclear TRIGA IPR-R1, do Centro de Desenvolvimento da Tecnologia Nuclear – CDTN, é um modelo Mark I, fabricado pela General Atomic de San Diego – Califórnia, refrigerado por água leve desmineralizada e tendo como combustível urânio enriquecido a 20% em ²³⁵U. Foi projetado para treinamento, pesquisa e ativação neutronica de materiais e produção de radioisótopos. Os reatores TRIGA (Training, Research, Isotopes, General Atomic) são caracterizados pela sua segurança intrínseca devido, principalmente, ao grande coeficiente negativo de temperatura/reatividade. Isto significa que um aumento da potência leva a um consequente aumento da temperatura da mistura combustível-moderador, causando o aparecimento de uma reatividade negativa que amortece gradualmente a taxa de aumento de potência e esta tende a se estabilizar. Outra característica de segurança dos reatores TRIGA é a alta retenção dos produtos de fissão no combustível, mesmo que o revestimento venha a sofrer falha, e um sistema passivo de remoção de calor no núcleo durante as operações.

O IPR-R1 é um reator nuclear de pesquisa do tipo piscina, refrigerado por circulação natural. Existe, entretanto, um sistema de refrigeração forçada do poço dotado de circuito primário e circuito secundário, que é utilizado para reduzir a temperatura do poço e, principalmente, para diminuir o nível de radiação ionizante na sala do reator. A Figura 1 mostra duas fotografias do poço e do núcleo do reator e a Figura 2 mostra um diagrama em corte do poço e a configuração do núcleo.

Fig. 1: Poço e núcleo do Reator Nuclear de Pesquisa TRIGA IPR-R1.

O núcleo do reator encontra-se atualmente carregado com 59 combustíveis com revestimento em alumínio e 5 combustíveis com revestimento em aço inoxidável. Um dos combustíveis com revestimento de aço possui em seu eixo central três termopares do tipo K (cronel/alumel). Este combustível instrumentado foi colocado no núcleo para possibilitar a realização de testes termohidráulicos, com o intuito de avaliar a performance do Reator IPR-R1 em operações em estado estacionário na nova potência máxima de 250 kW [1]. Realizaram-se várias medidas de temperaturas com o combustível instrumentado posicionado nos vários anéis do núcleo e em diferentes níveis de potências. Como resultado dos experimentos três núcleos novos processos de medida de potência por meios térmicos foram desenvolvidos, possibilitando o acompanhamento em tempo real da potência fornecida pelo núcleo.

CANAIS DE MEDIDA DE POTÊNCIA UTILIZANDO MÉTODOS NEUTRÓNICOS

A monitoração da potência dos reatores nucleares é sempre feita por meio de detectores nucleares os quais são calibrados por meios térmicos. No Reator IPR-R1 existem quatro câmaras sensíveis aos nêutrons posicionadas em torno do núcleo para medidas de fluxo neutrônico. O tipo de câmara usada e sua posição com relação ao núcleo determina a faixa de fluxos de nêutrons medidos, conforme descrito a seguir:

- Canal de partida; consiste de uma câmara de fissão (²³⁵U) com um amplificador de pulso que alimenta um circuito com indicador logarítmico, monitorando a evolução da taxa de nêutrons na partida do reator desde o nível da fonte até uns poucos watts (corresponde a uma taxa de contagem de 1 a 10⁵ cps).

- Canal logarítmico; consiste de uma câmara de ionização compensada que alimenta um amplificador logarítmico. O sinal vai para os mediadores de potência que indicam desde aproximadamente 0,1 W até a potência máxima em escala
logarítmica cobrindo nove décadas (de $10^{-3}$ a $10^8$ W). As indicações da velocidade do crescimento neutrónico (período) e a reatividade do sistema também provêm deste canal.

- Canal linear, consiste de uma câmara de ionização compensada que alimenta um amplificador linear. O sinal vai para comutador de escalas ($10^{-3}$ a $10^5$ W) onde se pode alterar a sensibilidade da medida, permitindo medir com precisão desde o nível da fonte até o nível de potência máxima.

- Canal percentual, consiste de uma câmara de ionização não-compensada que envia o sinal aos indicadores de potência calibrados em porcentagem (0 a 120%) da potência máxima.

Fig. 2: Vista em corte do poço e diagrama do núcleo do Reator TRIGA IPR-R1.

A instrumentação nuclear é utilizada para detectar os núcleons na partida do reator (multiplicação subcrítica) e após alcançar a criticalidade monitorando a variação do fluxo neutrónico para permitir o controle automático da reatividade e manter a potência estável.

Infelizmente as câmaras de ionização detectam o fluxo de núcleons termalizados apenas em sua vizinhança. Este sinal nem sempre é proporcional ao fluxo integral no núcleo e consequentemente à potência gerada. Além disso a resposta de cada detector nuclear é sensível a mudança na configuração do núcleo e, principalmente, na posição das barras de controle. Isto é importante nos reatores TRIGA por não terem um absorvedor de reatividade distribuído na água do núcleo e o controle da reatividade e a manutenção da criticalidade é feita pela inserção de barras de controle [2].

**CANAIS DE MEDIDA DE POTÊNCIA UTILIZANDO PROCESSOS TÉRMICOS**

**Medida de potência pelo balanço térmico**

O núcleo do reator é refrigerado pela convecção natural da água desmineralizada do poço. O calor é removido da água do poço e liberado para o ambiente por meio de um circuito de refrigeração dotado de circuito primário, circuito secundário e torre de refrigeração (Fig. 3). A temperatura da água do poço depende da potência de operação do reator assim como da temperatura ambiente, devido à dissipação de calor na torre de refrigeração. A potência total dissipada é determinada fazendo-se o balanço térmico do refrigerante nas entradas e saídas dos circuitos primário e
secundário e avaliando as fugas de calor. Estas perdas de calor representam uma pequena fração da potência total (cerca de 1,5% da potência total), conforme descrito em na referência [3].

![Diagrama de Circuito de Refrigeração Forçada do Reator TRIGA IPR-R1](image)

Fig. 3: Circuito de refrigeração forçada do Reator TRIGA IPR-R1.

As temperaturas de entrada e saída do refrigerante são medidas por quatro termoresistores de platina (PT-100), posicionados na entrada e saída das tubulações dos circuitos primário e secundário. A vazão da água do primário é medida pela queda de pressão em uma placa de orifício acoplada a um transmissor diferencial de pressão. No circuito secundário a vazão é medida por um rotametro. O transmissor de pressão foi calibrado e a equação de ajuste obtida foi adicionada ao programa do sistema de aquisição de dados. As linhas de medidas de temperatura foram calibradas como um todo, incluindo sensores, cabos, cartões de aquisição de dados e computador. As equações ajustadas também foram adicionadas ao programa de aquisição de dados.

A potência dissipada no circuito de refrigeração será o maior próximo da potência do reator, quanto mais próximo a temperatura da água do poço estiver da temperatura do meio ambiente. O estado estacionário é alcançado depois de algumas horas de operação do reator, quando então a potência dissipada no circuito de refrigeração adicionada às perdas, será igual a potência do núcleo.

A incerteza na medida de potência considera a propagação das incertezas de todos os parâmetros primários, conforme a metodologia descrita na referência [4]. A incerteza é calculada somente para a potência dissipada no circuito primário pois esta é atualmente o método padrão de medida de potência para o Reator TRIGA IPR-R1 [5].

A potência térmica dissipada nos circuitos primário e secundário são obtidas através do balanço térmico dado pela equação:

\[ q_{diss} = \dot{m} \cdot c_p \Delta T \quad (1) \]

Onde \( q_{diss} \) é a potência dissipada em cada circuito de refrigeração, \( \dot{m} \) é a vazão do refrigerante nos circuitos, \( c_p \) é o calor específico do refrigerante (água), e \( \Delta T \) a diferença entre as temperaturas de entrada e saída nos circuitos.

O programa do sistema de aquisição de dados calcula as potências dissipadas nos circuitos primário e secundário com os parâmetros coletados utilizados na Equação (1), com os valores de \( \dot{m} \) e \( c_p \) corrigidos em função da temperatura do refrigerante [6].

Para o cálculo das fugas de calor, utiliza-se uma termopar tipo K posicionado no centro do poço para a medida da temperatura média da água. Um termopar tipo K foi posicionado a cerca de 20 cm acima da superfície do poço.
para a medida da temperatura do ar na sala do reator. Dois termopares do tipo K foram distribuídos a 3 m de profundidade em torno do poço através de furos existentes nos pisos, para a medida da temperatura do solo. O núcleo do Reator IPR-R1 situa-se no fundo de um poço cilíndrico com 6,625 m de profundidade e 1,92 m de diâmetro. A superfície do poço situa-se a 25 cm abaixo do piso da sala. A água do poço do reator transfere calor para o meio ambiente por condução para solo através das paredes laterais e do fundo do poço e por convecção e evaporação para o ar da sala através da superfície da água. Todas estas perdas são calculadas pelo programa de aquisição de dados conforme descrito na referência [3].

A Figura 4 mostra a evolução da potência nos circuitos primário e secundário durante uma operação do reator. A Tabela 1 apresenta os resultados do balanço térmico desta operação e alguns dados experimentais.

![Fig. 4: Evolução das potências dissipadas nos circuitos de refrigeração primário e secundário.](image)

### Tabela 1: Balanço térmico do Reator TRIGA IPR-R1

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Vazão média do refrigerante no circuito primário</td>
<td>32,7 ± 0,4 m³/h</td>
</tr>
<tr>
<td>Temperatura média do refrigerante na entrada do primário</td>
<td>41,7 ± 0,3 °C</td>
</tr>
<tr>
<td>Temperatura média do refrigerante na saída do primário</td>
<td>34,8 ± 0,3 °C</td>
</tr>
<tr>
<td>Potência dissipada no circuito primário</td>
<td>261 kW</td>
</tr>
<tr>
<td>Perdas térmicas na piscina do reator</td>
<td>3,8 kW</td>
</tr>
<tr>
<td><strong>Potência térmica do reator</strong></td>
<td><strong>265 kW</strong></td>
</tr>
<tr>
<td>Desvio padrão</td>
<td>3,7 kW</td>
</tr>
<tr>
<td>Incerteza na medida da potência térmica</td>
<td>±19 kW (±7,2%)</td>
</tr>
<tr>
<td>Potência dissipada no circuito secundário</td>
<td>248 kW</td>
</tr>
</tbody>
</table>

### Medida de potência pela medida da temperatura do combustível e do poço

Um combustível instrumentado foi colocado no núcleo para experimentos termohidráulicos de avaliação da performance do Reator IPR-R1 [1]. O combustível instrumentado é igual ao combustível com revestimento de inox mas é equipado com três termopares de chromel-alumel fixados em seu eixo central de zircônio. As juntas quentes dos termopares estão localizadas no centro e espaçadas longitudinalmente em 25,4 mm entre si. A Figura 5 mostra em a) o elemento combustível instrumentado antes de ser posicionado no núcleo e em b) pode-se ver o elemento combustível instrumentado posicionado no anel B do núcleo. Na Figura 6 tem-se um diagrama e um desenho do combustível instrumentado. Na Tabela 2 encontram-se alguns dados deste elemento combustível [7].
Fig. 5: a) Elemento combustível instrumentado. b) Vista superior do núcleo com o elemento combustível instrumentado posicionado no anel B.

Fig. 6: Elemento combustível instrumentado.
<table>
<thead>
<tr>
<th>Parâmetro</th>
<th>Valor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Comprimento aquiccido</td>
<td>38,1 cm</td>
</tr>
<tr>
<td>Diâmetro externo</td>
<td>3,76 cm</td>
</tr>
<tr>
<td>Área externa ativa do elemento combustível</td>
<td>450,05 cm²</td>
</tr>
<tr>
<td>Área externa ativa do combustível (U-ZrH₂,a)</td>
<td>434,49 cm²</td>
</tr>
<tr>
<td>Volume ativo do elemento combustível</td>
<td>423,05 cm³</td>
</tr>
<tr>
<td>Volume do combustível (U-ZrH₂,a)</td>
<td>394,30 cm³</td>
</tr>
<tr>
<td>Potência (total no núcleo = 265 kW)</td>
<td>4,518 kW</td>
</tr>
</tbody>
</table>

Durante os experimentos termohidráulicos observou-se que a diferença entre a temperatura do elemento combustível e a água da piscina abaixo do núcleo do reator (temperatura de entrada do circuito primário de refrigeração, Fig. 3) mantinha-se constante, como pode ser observado na Fig. 7 para a potência de 265 kW.

![Diagrama](image)

Fig. 7: Evolução das temperaturas do combustível e da água abaixo do núcleo na potência de 265 kW.

Com o elemento combustível instrumentado posicionado na posição B1 do núcleo, a potência medida pelo canal neutrônico linear (com a indicação corrigida pela calibração de potência realizada) foi plotada em função da diferença de temperatura entre o combustível (média dos três termopares) e a temperatura de entrada do primário. Encontrou-se o seguinte polinômio relacionando os dois valores:

$$q = 2 \cdot 10^{-3} (\Delta T)^2 - 0.0045(\Delta T)^2 + 0.7666 \Delta T - 2.4475$$

onde:

- $q$ é a potência térmica calibrada do reator em [kW]
- $\Delta T$ é diferença entre a temperatura média fornecida pelos termopares do combustível instrumentado e a temperatura de entrada do primário, em [°C].

O coeficiente de determinação obtido foi um ($R^2 = 1$). A Equação (2) foi incluída no programa de aquisição de dados e este novo canal de medida de potência encontra-se disponível atualmente no Reator TRIGA IPR-R1. Após os experimentos o elemento combustível instrumentado foi mantido na posição B1 do núcleo e monitora a potência do reator e a temperatura do combustível em todas as operações. A Figura 8 compara um resultado da medida de potência utilizando o canal neutrônico linear e utilizando o canal de medida de potência pelo método da temperatura do combustível. Pode-se notar no gráfico um atraso na resposta no segundo canal devido a inércia térmica.
O limite de temperatura em estado estacionário, definido no Relatório de Análise de Acidentes do Reator TRIGA IPR-R1 [5], é de 550 °C. Baseando-se apenas neste limite operacional encontra-se, utilizando a Eq. (2), uma potência acima de 1 MW para o reator alcançar esta temperatura no combustível.

Na Figura 9 tem-se uma das interfaces gráficas que podem ser visualizadas no monitor de vídeo do sistema de aquisição de dados do Reator TRIGA IPR-R1 [8]. Esta tela consolida, em tempo real, as informações dos canais de medida de potência do reator, ou sejam, os canais neutrônicos e os três novos canais que utilizam processos térmicos.
CONCLUSÃO


Os métodos de medida de potência pelo balanço térmico e pela medida da temperatura do combustível são bastante precisos, mas não podem serem utilizados para monitorar os transições de potência. Para os transientes o monitoramento é feito pelos detectores neutrônicos, sendo estes calibrados pelo método do balanço térmico. Por outro lado, a resposta de cada detector nuclear é sensível às mudanças na configuração do núcleo, principalmente na posição das barras de controle. Os reatores de pesquisa não têm um absorvedor de nêutrons distribuídos na água de refrigeração para o controle da reatividade e o modo normal de obtenção e manutenção da criticalidade é pela inserção das barras de controle, que deixa o fluxo de nêutrons deformado no núcleo. No método de medida de potência pela temperatura do combustível, o aquecimento dos termopares devido à radiação gama pode ser desprezado devido a sua pequena massa e ao bom equilíbrio térmico entre os termopares e o combustível em sua volta.

AGRADECIMENTOS

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UNIDADES E NOMENCLATURA

$c_p$ calor específico à pressão constante (kJ/kg K)
$\dot{m}$ vazão de massa (kg/s)
$c$ centro do combustível (adimensional)
$p$ potência, taxa de transferência de calor (W)
$R_2$ coeficiente de determinação (adimensional)
$sar$ refrigeração (adimensional)
$T$ temperatura (°C, K)
$\Delta$ diferença (adimensional)
CONSTRUCTION OF THE MONITORING, PROCESSING AND LOGGING SYSTEMS FOR THE IPR-R1 TRIGA NUCLEAR REACTOR

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ABSTRACT

The actual supervisory program used in the IPR-R1 TRIGA Reactor was developed with the software VisiDAQ made by Advantech Co. This program version was doing in 1994 and ever since it has not been modernized. This paper presents the new supervisory program that has been designed in CDTN/CNEN, that using the software LabView by National Instruments in the Microsoft Windows XP platform. It will replace the old data acquisition system for a modern, user-friendly software, easy-to-maintain control system in a reasonable amount of time and cost. Reactor information could be transmitted via a high-speed Ethernet link from the Control Console System to the operator room or to a web browser of others computers.

1. INTRODUCTION

The IPR-R1 TRIGA Research Reactor, located at the Nuclear Technology Development Center - CDTN, has been operating for 44 years. The operational parameters are monitored and displayed by analog meters located at reactor console. The reactor operators registered the most important operational parameters manually in a logbook. This process is quite useful, but it can involve some human errors, parallax error, time delay, etc. It is also impossible for the operators to take notes of all variables involving the process mainly during fast power transients in some operations. Due to the recent experiments on thermohydraulics and reactor power calibrations [1] [2] and [3], it was necessary the development of a data acquisition system to make possible the tests. This system [4] still in use, is obsolete, due to the fast development in computer science and instrumentation. Therefore, it has been developed a new supervisory program using LabVIEW version 7.1 by National Instruments [5], in the Microsoft Windows XP Professional platform.

The LabVIEW software offers direct control of the hardware on the Data Acquisition (DAQ) board still in service [6]. For 20 years engineers and scientists have been using LabVIEW that is a powerful graphical development environment, for signal acquisition, measurement analysis, and data presentation. LabVIEW also provides the flexibility of a programming language without the complexity of traditional development tools. It is programmed with a set of graphical icons which are connected with "wires". The combination of a DAQ board and LabVIEW software makes a virtual instrument or a "VI". A "VI" can perform like an instrument and is programmable by the software with the advantage of logging data flexibility that is being measured.
2. HARDWARE DESCRIPTION

Three compensated ion chambers were located around the core, the detector's signals were amplified in a voltage amplifier. The amplified output voltage is between 0 to ±10V which is in turn the input sign of the Multiplexing Board and to the Analog to Digital Converter card of the PC with the associated software. The other analog signs collected by the data acquisition system are outputs of the reactor control console and from some digital indicators or directly from the thermocouples. Three input conditioning cards address these signs to an analog/digital converter card, which is installed in one Pentium 4 computer. The measure data are shown in the 19' computer video monitor. The main components of the instrumentation are described in the next sections.

3. DATA ACQUISITION CARDS

3.1. Amplifier and Multiplexing Cards

The analogical signs are received in three cards model PCLD-789 by Advantech Co [6] connected in cascade (Fig. 1), each one with 16 channels which totalize 48 inputs. These cards prepare the signs amplifying and filtering the noises and make the connection for a unique analogical output (multiplex action). One of the cards receives the signs directly from the thermocouples (range of ±100 mV). This card has a sensor that measures the temperature and makes the compensation of the cold junction adjusting the measured value. The other cards receive the signs from the control console (range of ±10 V). The main characteristics of the conditioning cards are:

- Accuracy: 0.0244% of the range ±1 LSB;
- Input: 16 differential channels;
- Input range: ±10 V maximum, varies with gain selection;
- Gain: 1, 2, 10, 50, 100, 200, 500 and 1000;
- Cold junction compensation: +24.4 mV/°C (0.0 V at 0.0 °C);

Table 1 shows the identifications of each signal collected by the cards. In the first column it is presented the number of the channel in each card. In the second column it is presented the identification (code) that is used in the data acquisition program to equation the collected sign, where: AI = Analog Input and TMP = Thermocouple. Finally, the third column shows the description and range which the collected signs were obtained.

![Diagram](image)

Figure 1. Data acquisition connection cards
Table 1. Operational parameter of the IPR-R1 TRIGA Reactor

<table>
<thead>
<tr>
<th>Channel</th>
<th>Input</th>
<th>Collected Sign</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>TMP 10</td>
<td>Water pool temperature (lower 1), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>1</td>
<td>TMP 1</td>
<td>Fuel temperature (upper), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>2</td>
<td>TMP 2</td>
<td>Fuel temperature (medium), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>3</td>
<td>TMP 3</td>
<td>Fuel temperature (lower), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>4</td>
<td>TMP 4</td>
<td>Air temperature above the pool, (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>5</td>
<td>TMP 5</td>
<td>Ground temperature around the pool (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>6</td>
<td>TMP 6</td>
<td>Outlet core channel temperature), (thermocouples, -6 to 55 mv)</td>
</tr>
<tr>
<td>7</td>
<td>TMP 7</td>
<td>Inlet core channel temperature (lower), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>8</td>
<td>TMP 8</td>
<td>Water pool temperature (upper 1), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>9</td>
<td>TMP 9</td>
<td>Water pool temperature (upper 2) (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>10</td>
<td>AI 21</td>
<td>Pool water temperature , (PT-100, 4 to 20 mA)</td>
</tr>
<tr>
<td>11</td>
<td>AI 4</td>
<td>Water inlet temperature in secondary circuit, (pt-100, 4 to 20 mA)</td>
</tr>
<tr>
<td>12</td>
<td>AI 5</td>
<td>Water outlet temperature in secondary circuit, (pt-100, 4 to 20 mA)</td>
</tr>
<tr>
<td>13</td>
<td>AI 1</td>
<td>Water flow in the primary circuit, (4 to 20mA)</td>
</tr>
<tr>
<td>14</td>
<td>AI 2</td>
<td>Water inlet temperature in primary circuit, pt-100, (4 to 20 mA)</td>
</tr>
<tr>
<td>15</td>
<td>AI 3</td>
<td>Water outlet temperature in primary circuit, pt-100, (4 to 20 mA)</td>
</tr>
<tr>
<td>16</td>
<td>-</td>
<td>Reserved to be used for air relative humidity</td>
</tr>
<tr>
<td>17</td>
<td>AI 6</td>
<td>Logarithmic channel power, (0 to 10 V)</td>
</tr>
<tr>
<td>18</td>
<td>AI 7</td>
<td>Linear channel power, (0 to 10 V)</td>
</tr>
<tr>
<td>19</td>
<td>AI 8</td>
<td>Percent power channel, (0 to 10 V)</td>
</tr>
<tr>
<td>20</td>
<td>AI 14</td>
<td>Period, (0 to 10 V)</td>
</tr>
<tr>
<td>21</td>
<td>AI 15</td>
<td>Reactivity, (-10V to +10 V)</td>
</tr>
<tr>
<td>22</td>
<td>AI 16</td>
<td>Start-up channel counter, (0 to 10 V)</td>
</tr>
<tr>
<td>23</td>
<td>AI 18</td>
<td>Safety rod position, (0 to 2.5 V)</td>
</tr>
<tr>
<td>24</td>
<td>AI 19</td>
<td>Shim rod position, (0 to 2.5 V)</td>
</tr>
<tr>
<td>25</td>
<td>AI 20</td>
<td>Regulating rod position, (0 to 2.5 V)</td>
</tr>
<tr>
<td>26</td>
<td>-</td>
<td>Aerosols radiation (disabled), (0 to 10 V)</td>
</tr>
<tr>
<td>27</td>
<td>AI 9</td>
<td>Well radiation, (0 to 10 V)</td>
</tr>
<tr>
<td>28</td>
<td>AI 10</td>
<td>Area radiation, (0 to 10 V)</td>
</tr>
<tr>
<td>29</td>
<td>AI 11</td>
<td>Primary circuit inlet radiation, (0 to 10 V)</td>
</tr>
<tr>
<td>30</td>
<td>AI 12</td>
<td>Ion changer radiation, (0 to 10 V)</td>
</tr>
<tr>
<td>31</td>
<td>AI 13</td>
<td>Inlet secondary circuit radiation, (0 to 10 V)</td>
</tr>
<tr>
<td>32</td>
<td>-</td>
<td>Reserve</td>
</tr>
<tr>
<td>33</td>
<td>TMP 11</td>
<td>Pool temperature (lower 2), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>34</td>
<td>TMP 12</td>
<td>Pool temperature (lower 3), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>35</td>
<td>TMP 13</td>
<td>Pool temperature (lower 4), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>36</td>
<td>TMP 14</td>
<td>Pool temperature (lower 2), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>37</td>
<td>TMP 15</td>
<td>Pool temperature (lower 3), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>38</td>
<td>TMP 16</td>
<td>Pool temperature (lower 4), (thermocouples, -6 to 55 mV)</td>
</tr>
<tr>
<td>39</td>
<td>TMP 17</td>
<td>Ground temperature around the pool (2) (thermocouples, -6 to 55 mv)</td>
</tr>
<tr>
<td>40 to 48</td>
<td>-</td>
<td>Reserve</td>
</tr>
</tbody>
</table>

3.2. Analog to Digital Conversion Card

The outputs of the three conditioning cards are addressed to the analog input plug of the data acquisition card, model PCL-818hd by Advantech Co [6]. This is a high-speed data transference card installed in the computer case, which transforms the analog input signs into digital sign. This card has the following main characteristics:
- Accuracy: 0.01% of the range ± 1 LSB;
- Resolution: 12 bits;
- Sampling rate: up to 100 kHz with DMA transfer;
- Over voltage: continuous ±30 V max.

4. DATA ACQUISITION SOFTWARE

The LabVIEW is programmed with set of icons that represents controls and functions, available in the menu of the software. Such a programming is called visual programming. The user interface which is called a VI (Virtual Instrument) consists of two parts - a front panel and a diagram. This is similar to that of an instrument where a front panel is used for an input, output controls, and to display the data whereas the circuit resides on the circuit board. Similarly it can bring the buttons, indicators and graphing and display functions on the front panel. The VI also calculates mean and standard deviation of the data and plots it. Alarms have been set up so that if the temperature falls below or above a certain set value the alarm goes ON. The LabVIEW can be used to perform system simulations, since it contains filters many commonly used, digital signal processing, and statistical functions.

The main indications of the control console are collected by the data acquisition system, including the positions of the three control rods. These signs come from the rack of instruments and from the reactor control console and they are input in channels 1 to 15 of Card 2. A description of all signs collected from the control console is not presented in this paper. It was accomplished all the answers of the parameters collected and the found equations were introduced in the data acquisition program to transform the signals of Volt into engineering units. The program presents four screens; two of them are shown in Fig. 2 (Start up Channel, Control Rods, Reactivity, Levels of Radiation). The others screens are: Power Channels / Cooling System and Temperatures.

4.1 Control, Start up Channel, Period and Reactivity

In this screen, which was shown in Fig. 1, the start up of the reactor can be accompanied through the neutron evolution counting rate. The positions of the three control rods of the reactor can be visualized in graphics of the rods or in digital indicators. Three graphics also show the evolution of the control rods position in the last 60 minutes. The reactivity of the reactor in [pcm] and in [dollar] is given by digital counters. This screen also shows the positive period of the reactor (T) in [s] and the start up rate (SUR) in [dpm]. A webcam films the reactor core and it shows the image in this screen.

4.2. Radiation Levels

The gamma radiation levels at the reactor area are measured in the following positions: in the Control Room (AEROSOSIS); at about 30 cm above the reactor pool (POÇO); at 2 m above the reactor pool (AREA); at the inlet piping of the primary cooling loop heat exchanger (ENTRADA PRIMÁRIO); in the ion exchanger system (RESINAS); and at the outlet piping of the secondary cooling loop heat exchanger (SAÍDA SECUNDÁRIO).

Figure 3 shows the screen of the radiation levels monitoring. The six radiation levels of the monitoring channels are shown in analog and digital indicators and graphics, and give the evolution of the radiation levels in the last 60 minutes.

INAC 2007, Santos, SP, Brazil.
4.3. Power Channels

This screen shows the evolution of the reactor powers supplied by three conventional neutron channels of measurement: Logarithmic Channel, Linear Channel and Percent Power Channel. The values are given by digital indication and by graphics that show the last 60 minutes. The evolution of the power dissipated in the primary and secondary-cooling systems is also shown. After several hours of reactor operation, when it is reached the thermal balance with the environment, the power of the reactor will be the closer to the power dissipated in the primary coolant loop and the thermal losses will be smaller. Those losses value are also indicated in the screen. The reactor power is monitored also by the increase of the temperature in the center of the instrumented fuel.

4.4. Cooling System and Temperatures

This screen shows all parameters of the primary and secondary cooling loops. The following signals are monitored and shown on the screen:

- The average value of the inlet and outlet temperatures of the primary and secondary loops and its standard deviations.
- The average value of the flow rate and its standard deviation.
- The power dissipated in the primary and secondary cooling loops.
- The average value of the temperatures in the reactor pool and its standard deviations.
- The temperature of the air above the reactor pool and in two points of the soil.
- The average value of the temperatures in the three thermocouples of the instrumented fuel element.
- The time elapsed from the program beginning in [s], [min] and [h].

5. CONCLUSIONS

The new data acquisition based in the microcomputer has been designed and developed to allow the real-time collection of all operational parameters and information from the IPR-R1 TRIGA Reactor. The operational parameters and information from the reactor console have been fed to the computer through an interface unit that includes hardware like multiplexer,
A/D converter and computer interface. Information on all aspects of reactor operation is displayed on the Control Console System. The two color graphics monitors can display real-time operations data in concise, accurate, and easily understood formats. Bar graph indicators and visual and audible annunciators are also provided. Information displayed on the monitor can be recorded on hard copy using the graphics printer in the Control Console System. The DAC collects data during reactor operations and stores it in a historical database. Reactor operations can then be replayed in real-time or slow motion. This record is a powerful tool that can be used for operations review and maintenance troubleshooting.

The system also has been useful to provide more information during the reactor operation and does not influence the original reactor measuring and control instrumentation by any way. The IPR-R1 Reactor isn’t controlled by the automatic way (startup, control rod movements and scrams). This operational philosophy was defined about 40 years ago, when the automation means were limited. Today, with the technology developed and with the use of a computer it’s possible to execute all nuclear power plant operations [7]. The TRIGA Reactor of the Pennsylvania University, in the USA and other research reactors, have systems coupled to a special program that controls the installation, from the start up at the start down [8]. It is suggested with this paper to initiate studies that permit to automatize all the IPR-R1 TRIGA Reactor operation.

REFERENCES


Experimental determination of heat transfer coefficients in uranium zirconium hydride fuel rod

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Abstract: The heat generated by nuclear fission is transferred from fuel elements to the cooling system through the fuel-to-cladding gap and the cladding to coolant interfaces. The fuel thermal conductivity and the heat transfer coefficient from the cladding to the coolant were evaluated experimentally. A correlation for the gap conductance between the fuel and the cladding was also presented. As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to sub cooled nucleate boiling. Results indicated that sub cooled boiling occurs at the cladding surface in the central channels of the reactor core at power levels in excess of 60 kW.

Keywords: heat transfer; nuclear fuel rod; natural convection; sub cooled nucleate boiling; instrumented fuel element; TRIGA reactor.

Reference: to this paper should be made as follows: Mesquita, A.Z.; Rezende, H.C. (2006), "Experimental determination of heat transfer coefficients in uranium zirconium hydride fuel rod", Int. J. Nuclear Energy Science and Technology, Vol. x, No. x, xxxx, pp. xxx-xxx

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1. Introduction

The IPR-R1 TRIGA Nuclear Research Reactor (Fig. 1), installed at Centre of Nuclear Technology Development (Brazil), is a pool type reactor cooled by natural circulation, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in $^{235}U$. The core contains 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements. One of these steel-clad fuel elements is instrumented in the centre with three thermocouples as shown in Fig. 1 and Fig. 2.

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Figure 1  Core upper view with the instrumented fuel element in ring B

Figure 2  Instrumented fuel element scheme
Experimental determination of heat transfer coefficients in uranium zirconium hydride fuel rod

The heat generated by fissions is transferred from fuel elements to the cooling system through a fuel-to-cladding interface and from cladding to coolant. The objective of the thermal and hydrodynamic projects of the reactors is to remove the heat safely, without producing excessive temperature in the fuel elements. The regions of the reactor core where boiling occurs at various power levels can be determined from the heat transfer coefficient data.

The thermal conductivity ($k$) of the metallic alloys is mainly a function of temperature. In nuclear fuels, this relationship is more complicated because $k$ also becomes a function of irradiation as a result of change in the chemical and physical composition (porosity changes due to temperature and fissile products). Many factors affect the fuel thermal conductivity. The major factors are temperature, porosity, oxygen to metal atom ratio, PuO$_2$ content, pellet cracking, and burnup. The second largest resistance to heat conduction in the fuel rod is due to the gap. Several correlations exist [1] to evaluate its value in power reactors fuels, which use mainly uranium oxide. The only reference found to TRIGA reactors fuel is General Atomic [2] that recommends the use of three hypotheses for the heat transfer coefficient in the gap. The heat transfer coefficient ($h$) is a property not only of the system but also depends on the fluid properties. The determination of $h$ is a complex process that depends on the thermal conductivity, density, viscosity, velocity, dimensions and specific heat. All these parameters are temperature-dependent and change when heat is being transferred from the heated wall to the fluid. An operational computer program and a data acquisition and signal processing system were developed as part of this research project [9] to allow on-line monitoring of the operational parameters.

1. Overall thermal conductivity of the fuel elements

From Fourier equation described in [3, 6], it was obtained the expression of overall thermal conductivity ($k_o$), in [W/m$^2$K], for cylindrical fuel elements

$$k_o = \frac{q''''r^2}{4(T_o - T_{sat})}$$  \hspace{1cm} (1)

where $q''''$ is the volumetric rate of heat generation [W/m$^3$]. $T_o$ and $T_{sat}$ are the fuel central temperature and the surface temperature [°C] and $r$ is the fuel element radius [m].

The temperature at the centre of the fuel was measured. The heat transfer regime at the power of 265 kW in all fuel elements is the subcooled nucleate boiling. The cladding outside temperature is the water saturation temperature ($T_{sat}$) at pressure of 1.5 bar (atmospheric pressure added up of the water column of 5.2 m), increased of the wall superheat ($\Delta T_{sat}$). The superficial temperature ($T_{sat}$) in [°C] is found using the expression below, where $T_{sat}$ is equal to 111 37 °C [7].

$$T_{sat} = T_{sat} + \Delta T_{sat}$$  \hspace{1cm} (2)

The wall superheat is obtained by using the correlation proposed by McAdams found in [8].

$$\Delta T_{sat} = 0.81(q'')^{0.237}$$  \hspace{1cm} (3)

with $q''$ in [W/m$^2$] and $T_{sat}$ in [°C].

A fuel element instrumented with three type K thermocouples was introduced into position B1 as shown in Fig. 1. Two thermocouples were also placed in two core channels adjacent to position B1.
2. Heat transfer in the reactor core

2.1 Single-Phase Region

The heat transfer coefficient in single-phase region \( h_{np} \) was calculated with the Dittus-Boelter correlation described in [10], valid for turbulent flow in narrow channels, given for:

\[
h_{np} = 0.023 \frac{k}{D_w} \left( \frac{G D_w}{\mu} \right)^{0.8} \left( \frac{c_p \mu}{k} \right)^{0.4}.
\]  

where \( D_w = \frac{4A}{P_w} \) is the hydraulic diameter of the channel based on the wetted perimeter, \( A \) is the flow area [m²], \( P_w \) is the wetted perimeter [m], \( G \) is the mass flow [kg/m²s], \( c_p \) is the isobaric specific heat [J/kgK], \( k \) is the thermal conductivity [W/mK], and, \( \mu \) is the fluid dynamic viscosity [kg/ms]. The fluid properties for the IPR-R1 TRIGA core are calculated for the bulk water temperature at 1.5 bar.

The two hottest channels in the core are Channel 0 and Channel 1' (Fig. 3). The heat transfer coefficient was estimated using the Dittus-Boelter correlation. The inlet and outlet temperatures in Channel 0 were considered as being the same as in Channel 1'. Table 1 gives the geometric data of Channel 0 and Channel 1' and the percent contribution of each fuel element to the channel power. The curves of single-phase heat transfer, as function of \( \Delta T_{net} \), are presented in the Fig. 4.

Figure 3  The two hottest channels in the core
Experimental determination of heat transfer coefficients in uranium zirconium hydride fuel rod

Figure 4  Heat-transfer regimes in the fuel element surface

The mass flow rate is given indirectly from the thermal balance along the channel using measurements of the water inlet and outlet temperatures:

\[ q = \dot{m}c_p\Delta T \]  \hspace{1cm} (5)

where \( q \) is the power supplied to the channel [W], \( \dot{m} \) is the mass flow rate in the channel [kg/s], \( c_p \) is the isobaric specific heat of the water [J/kg K]; and, \( \Delta T \) is the temperature difference along the channel [°C].

Table 1  Channel 0 and Channel 1' Characteristics [9]

<table>
<thead>
<tr>
<th></th>
<th>Channel 0</th>
<th>Channel 1'</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Area (A)</td>
<td>1.574</td>
<td>8.214</td>
<td>cm²</td>
</tr>
<tr>
<td>Wetted Perimeter (P_w)</td>
<td>5.901</td>
<td>17.643</td>
<td>cm</td>
</tr>
<tr>
<td>Heated Perimeter (P_h)</td>
<td>3.906</td>
<td>15.156</td>
<td>cm</td>
</tr>
<tr>
<td>Hydraulic Diameter (D_h)</td>
<td>1.067</td>
<td>1.862</td>
<td>cm</td>
</tr>
<tr>
<td>B1 and C1 Fuel Diameter (stainless)</td>
<td>3.76</td>
<td>3.76</td>
<td>cm</td>
</tr>
<tr>
<td>B6 and C12 Fuel Diameter (Al)</td>
<td>3.73</td>
<td>3.73</td>
<td>cm</td>
</tr>
<tr>
<td>C1 Control Rod Diameter</td>
<td>3.80</td>
<td>3.80</td>
<td>cm</td>
</tr>
<tr>
<td>Central Thimble</td>
<td>3.81</td>
<td>3.81</td>
<td>cm</td>
</tr>
<tr>
<td>Core Total Power (265 kW)</td>
<td>100</td>
<td>100</td>
<td>%</td>
</tr>
<tr>
<td>B1 Fuel Contribution</td>
<td>0.54</td>
<td>1.11</td>
<td>%</td>
</tr>
<tr>
<td>B6 Fuel Contribution</td>
<td>0.46</td>
<td>0.94</td>
<td>%</td>
</tr>
<tr>
<td>C11 Fuel Contribution</td>
<td>-</td>
<td>0.57</td>
<td>%</td>
</tr>
<tr>
<td>C12 Fuel Contribution</td>
<td>-</td>
<td>1.08</td>
<td>%</td>
</tr>
<tr>
<td>Total Power of the Channel</td>
<td>1.00</td>
<td>3.70</td>
<td>%</td>
</tr>
</tbody>
</table>
A. Z. Mesquita and H. C. Rezende

The reactor was operated on steps of about 50 kW until 205 kW and data were collected in function of the power supplied to Channel 1 and Channel 0. The values of the water thermodynamic properties at the pressure 1.5 bar as function of the bulk water temperature at the channel were taken from Wagner and Krane [7]. The curve for heat transfer coefficient \( h_{sw} \) in the single-phase region is shown in Fig. 5 as function of the power.

### 2.2 Sub cooled Nucleate Boiling Region

For the sub cooled nucleated boiling region (local or surface boiling), the expression used is shown below, according to [11, 12]:

\[
h_{sw} = q'' \div \Delta T_{sw}
\]

where \( h_{sw} \) is the convective heat-transfer coefficient from the fuel cladding outer surface to the water \([\text{W/m}^2\text{K}]\); \( q'' \) is the fuel surface heat flux \([\text{W/m}^2]\); and, \( \Delta T_{sw} \) is the surface superheat in contact with the water \([\text{°C}]\).

Figure 4 presents the fuel element surface heat transfer coefficient for the coolant as a function of the superheat, in both regimes. This curve is specific for the IPR-R1 TRIGA reactor conditions. The correlation used for the sub cooled nucleate boiling is not valid for single-phase convection region, as well as the Dittus-Boelter correlation is not valid for the boiling region. The transition point between single-phase convection regime to sub cooled nucleate boiling regime (onset of nucleate boiling) is approximately 60 kW as shown in the graph.

Figure 5 presents the curves for the heat transfer coefficient \( h_{sw} \) on the fuel element surface and for the overall thermal conductivity \( k_g \) in fuel element as function of the power, obtained for the instrumented fuel at core position B1.

**Figure 5** Overall fuel element thermal conductivity and cladding heat transfer coefficient to the coolant
3. Heat transfer coefficient in the fuel gap

The instrumented fuel element is composed of a central zirconium filler rod with 6.25 mm in diameter, the active part of the fuel, formed by uranium zirconium hydride alloy (U-ZrH₁₋₅), an interface (gap) between the fuel and the cladding, and the 304 stainless steel cladding. The thermocouples are fixed in the central rod. It is assumed that all heat flux is in the radial direction. Using the analogy with electric circuits, the resistance to the heat conduction from the fuel centre to the coolant \( R_D \) is given by the sum of the fuel components resistances.

The fuel element configuration is shown in Fig. 7. The axial heat conduction and the presence of the central pin of zirconium were not considered. The thermal conductivity of the U-ZrH₁₋₅ fuel is given by (13):

\[
k_{UZH} = 0.0073T + 17.58,
\]

with \( T \) in [°C] and \( k_{UZH} \) in [W/mK]. The thermal conductivity of the AISI 304 steel cladding is given by (14):

\[
k_{c} = 3.17 \times 10^7T^2 - 6.67 \times 10^5T^2 + 1.81 \times 10^5T + 14.46,
\]

with \( T \) in [°C] and \( k_c \) in [W/mK].

The value of \( R_{gap} \) is the value of the overall resistance of the fuel element \( (R_D) \) less the values of other component resistance. It is found with the values of \( k_e \) and \( k_{c} \) obtained previously and with the values of \( k \) for the fuel alloy and for the cladding corrected in function of temperature. The heat transfer coefficient in the gap is:

\[
h_{gap} = \frac{2}{R_D} \left( \frac{k_e k_{UZH} k_c}{k_{UZH} k_c - k_e k_{UZH} - 2k_e k_{UZH} \ln(r_2 \div r_1)} \right)
\]

The graph of heat transfer coefficient through the gap is shown in Fig. 7, as a function of the reactor power. This figure also shows three theoretical values recommended by General Atomic for the heat transfer coefficient [2].

Figure 6  Heat transfer coefficient through the gap as a function of the power

![Graph of heat transfer coefficient through the gap as a function of the power](image)
A. Z. Mesquita and H. C. Rezende

Figure 7 Fuel element configuration

4. Fuel rod temperature profile

From the temperature in the center of the fuel and using the equations of conduction for the fuel element geometry, it is possible to obtain the radial temperature distribution in the fuel element. Figure 8 shows the experimental radial profile of maximum fuel temperature in position B1 and it is compared with the PANTERA code results [15]. The instrumented fuel element was used to measure the fuel temperature at several reactor powers. The results are also shown in Fig. 9.

Figure 8 Experimental fuel rod radial temperature profile in position B1 at 265 kW
5. Conclusion

Subcooled pool boiling occurs above approximately 60 kW on the cladding surface in the central channels of the IPR-R1 TRIGA core. However, the high heat transfer coefficient due to subcooled boiling causes the cladding temperature to be quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. The IPR-R1 TRIGA Reactor normally operates in the range from 100 kW until a maximum of 250 kW. On these power levels the heat transfer regime between the clad surface and the coolant is subcooled nucleate boiling in the hottest fuel element. Boiling heat transfer is usually the most efficient heat transfer pattern in nuclear reactors core [5]. Another important aspect of the reactor operation safety is that it is far from the occurrence of critical heat flux [16].

Acknowledgments

The authors thank to the operation staff of the IPR-R1 TRIGA Reactor for their help during the experiments.

References

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Experimental Heat Transfer Analysis of the IPR-R1 TRIGA Reactor

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Abstract. The 250 kW IPR-R1 TRIGA Nuclear Research Reactor, installed at Nuclear Technology Development Center (CDTN) in Belo Horizonte, Brazil, is a pool type reactor cooled by natural circulation, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in $^{235}$U. The heat generated by nuclear fission is transferred from fuel elements to the cooling system through the fuel-to-cladding gap and the cladding to coolant interfaces. The fuel thermal conductivity and the heat transfer coefficient from the cladding to the coolant were evaluated experimentally. A correlation for the gap conductance between the fuel and the cladding was also presented. As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to subcooled nucleate boiling. Results indicated that subcooled boiling occurs at the cladding surface in the central channels of the reactor core at power levels above approximately 60 kW. However, the high heat transfer coefficient due to subcooled boiling causes the cladding temperature be quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. An operational computer program and a data acquisition and signal processing system were developed as part of this research project to allow on line monitoring of the operational parameters.

INTRODUCTION

The IPR-1 TRIGA core contains 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements. One of these steel-clad fuel elements is instrumented in the center with three thermocouples. Figure 1 shows the core upper view with the instrumented fuel element in ring B.

FIG. 1. Pool and core top view with the instrumented fuel element in ring B

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The objective of the thermal and hydrodynamic projects of the reactors is to remove the heat safely, without producing excessive temperature in the fuel elements. The regions of the reactor core where boiling occurs at various power levels can be determined from the heat transfer coefficient data.

The thermal conductivity \((k)\) of the metallic alloys is mainly a function of temperature. In nuclear fuels, this relationship is more complicated because \(k\) also becomes a function of irradiation as a result of change in the chemical and physical composition (porosity changes due to temperature and fission products). Many factors affect the fuel thermal conductivity. The major factors are temperature, porosity, oxygen to metal atom ratio, PuO\(_2\) content, pellet cracking, and burnup. The second largest resistance to heat conduction in the fuel rod is due to the gap. Several correlations exist [1] to evaluate its value in power reactors fuels, which use mainly uranium oxide. The only reference found to TRIGA reactors fuel is General Atomic [2] that recommends the use of three hypotheses for the heat transfer coefficient in the gap. The heat transfer coefficient \((h)\) is a property not only of the system but also depends on the fluid properties. The determination of \(h\) is a complex process that depends on the thermal conductivity, density, viscosity, velocity, dimensions and specific heat. All these parameters are temperature-dependent and change when heat is being transferred from the heated wall to the fluid. An operational computer program and a data acquisition and signal processing system were developed as part of this research project [3] to allow on-line monitoring of the operational parameters.

As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to subcooled nucleate boiling. Results indicated that subcooled boiling occurs at the cladding surface in the central channels of the reactor core at power levels in excess of 60 kW [4].

**OVERALL THERMAL CONDUCTIVITY OF THE FUEL ELEMENTS**

From Fourier equation described in [5, 6], it was obtained the expression of overall thermal conductivity \((k)\), in [W/mK], for cylindrical fuel elements

\[
k_x = \frac{q'' r^2}{4(T_a - T_{sat})}, \tag{1}
\]

where \(q''\) is the volumetric rate of heat generation [W/m\(^3\)], \(T_a\) and \(T_{sat}\) are the fuel central temperature and the surface temperature [°C] and \(r\) is the fuel element radius [m].

The temperature at the center of the fuel was measured. The heat transfer regime at the power of 265 kW in all fuel elements is the subcooled nucleate boiling. The cladding outside temperature is the water saturation temperature \((T_{sat})\) at the pressure of 1.5 bar (atmospheric pressure added up of the water column of \(\sim 5.2\) m), increased of the wall superheat \((\Delta T_{sat})\). The superficial temperature \((T_{sat})\) in [°C] is found using the expression below, where \(T_{sat}\) is equal to 111.37 °C [7].

\[
T_{sat} = T_{sat} + \Delta T_{sat}. \tag{2}
\]

The wall superheat is obtained by using the correlation proposed by McAdams found in [8],

\[
\Delta T_{sat} = 0.81(q'' r^2)^{0.259}, \tag{3}
\]

with \(q''\) in [W/m\(^3\)] and \(T_{sat}\) in [°C].

A fuel element instrumented with three type K thermocouples was introduced into position B1 of the core. Figure 2 shows the instrumented fuel element and the TRIGA IPR-R1 core. Two thermocouples were also placed in two core channels adjacent to position B1.
HEAT TRANSFER IN THE REACTOR CORE

Single-Phase Region

The heat transfer coefficient in single-phase region \( (h_w) \) was calculated with the Dittus-Boelter correlation described in [9] valid for turbulent flow in narrow channels, given for:

\[
h_w = 0.023 \frac{k}{D_w} \left( \frac{GD_w}{\mu} \right)^{0.8} \left( \frac{c_p \mu}{k} \right)^{0.4},
\]

where \( D_w = 4A/P_w \) is the hydraulic diameter of the channel based on the wetted perimeter; \( A \) is the flow area \([\text{m}^2]\); \( P_w \) is the wetted perimeter \([\text{m}]\); \( G \) is the mass flow \([\text{kg/m}^3\text{s}]\); \( c_p \) is the isobaric specific heat \([\text{J/kgK}]\); \( k \) is the thermal conductivity \([\text{W/mK}]\); and, \( \mu \) is the fluid dynamic viscosity \([\text{kg/ms}]\). The fluid properties for the IPR-R1 TRIGA core are calculated for the bulk water temperature at 1.5 bar.

The two hottest channels in the core are Channel 0 and Channel 1' (Fig. 3). The heat transfer coefficient was estimated using the Dittus-Boelter correlation. In the top gride plate above the Channel
Amir Zacarias Mesquita

1' there is a hole to insert thermocouples. Above the Channel 0 there isn’t hole. The inlet and outlet temperatures in Channel 0 were considered as being the as in Channel 1'. Table 1 gives the geometric data of Channel 0 and Channel 1' and the percent contribution of each fuel element to the channel power. The curves of single-phase heat transfer, as function of $\Delta T_{sat}$, are presented in the Fig. 4.

**FIG. 3. The two hottest channels in the core**

![Diagram showing the hottest channels in the core](image)

**FIG. 4. Heat-transfer regimes in the fuel element surface of the IPR-R1 TRIGA Reactor**

The mass flow rate is given indirectly from the thermal balance along the channel using measurements of the water inlet and outlet temperatures:

$$ q = m c_p \Delta T, $$

(5)
Amr Zacarias Mesquita

where $q$ is the power supplied to the channel [kW]; $\dot{m}$ is the mass flow rate in the channel [kg/s]; $c_p$ is the isobaric specific heat of the water [J/kgK]; and, $\Delta T$ is the temperature difference along the channel [°C].

| Table 1. Channel 0 and Channel 1' Characteristics [4] |
|-----------------|-----------------|------------------|
| Area (A)        | 1.574           | 8.214            |
| Wetted Perimeter (P_w) | 5.901       | 17.643           |
| Heated Perimeter (P_h) | 3.906       | 15.156           |
| Hydraulic Diameter (D_w) | 1.067       | 1.862            |
| B1 and C1 Fuel Diameter (stainless) | 3.76          | 3.76             |
| B6 and C12 Fuel Diameter (Al) | 3.73          | 3.73             |
| C1 Control Rod Diameter | 3.80          | 3.80             |
| Central Thimble   | 3.81            | 3.81             |
| Core Total Power (265 kW) | 100          | 100              |
| B1 Fuel Contribution | 0.54        | 1.11             |
| B6 Fuel Contribution | 0.46        | 0.94             |
| C11 Fuel Contribution | -           | 0.57             |
| C12 Fuel Contribution | -           | 1.08             |
| Total Power of the Channel | 1.00        | 3.70             |

The reactor was operated on steps of about 50 kW until 265 kW and data were collected in function of the power supplied to Channel 1' and Channel 0. The values of the water thermodynamic properties at the pressure 1.5 bar as function of the bulk water temperature at the channel were taken from Wagner and Kruse [9]. The curve for heat transfer coefficient ($h_{mr}$) in the single-phase region is shown in Fig. 5 as function of the power.

**Subcooled Nucleate Boiling Region**

For the subcooled nucleate boiling region (local or surface boiling), the expression used is shown below, according to [10, 11]:

$$h_{mr} = \frac{q'}{\Delta T_{mr}}.$$  \hspace{5cm} (6)

where $h_{mr}$ is the convective heat-transfer coefficient from the fuel cladding outer surface to the water [kW/m²K]; $q'$ is the fuel surface heat flux [kW/m²]; and, $\Delta T_{mr}$ is the surface superheat in contact with the water [°C].

Figure 4 presents the fuel element surface heat transfer coefficient for the coolant as a function of the superheat, in both regimes. This curve is specific for the IPR-R1 TRIGA reactor conditions. The correlation used for the subcooled nucleate boiling is not valid for single-phase convection region, as well as the Dittus-Boelter correlation is not valid for the boiling region. The transition point between single-phase convection regime to subcooled nucleate boiling regime (onset of nucleate boiling) is approximately 60 kW as shown in the graph.

Figure 5 presents the curves for the heat transfer coefficient ($h_{mr}$) on the fuel element surface and for the overall thermal conductivity ($k_f$) in fuel element as function of the power, obtained for the instrumented fuel at core position B1.
HEAT TRANSFER COEFFICIENT IN THE FUEL GAP

The instrumented fuel element is composed of a central zirconium filler rod with 6.25 mm in diameter, the active part of the fuel, formed by uranium zirconium hydride alloy (U-ZrH$_{1.6}$), an interface (gap) between the fuel and the cladding, and the 304 stainless steel cladding. The thermocouples are fixed in the central rod. It is assumed that all heat flux is in the radial direction. Using the analogy with electric circuits, the resistance to the heat conduction from the fuel center to the coolant ($R_p$) is given by the sum of the fuel components resistances.

The fuel element configuration is shown in Fig. 6. The axial heat conduction and the presence of the central pin of zirconium were not considered. The thermal conductivity of the U-ZrH$_{1.6}$ fuel is given by [12]:

$$k_{UZH} = 0.0075 T + 17.58,$$

with $T$ in [°C] and $k_{UZH}$ in [W/mK]. The thermal conductivity of the AISI 304 steel cladding is given by [13]:

$$k_{rev} = 3.17 \times 10^{-9} T^2 - 6.67 \times 10^{-6} T^2 + 1.81 \times 10^{-2} T + 14.46,$$

with $T$ in [°C] and $k_{rev}$ in [W/mK].

The value of $R_{gap}$ is the value of the overall resistance of the fuel element ($R_p$) less the values of other component resistance. It is found with the values of $k_g$ and $h_{sur}$ obtained previously and with the values of $k$ for the fuel alloy and for the cladding corrected in function of temperature. The heat transfer coefficient in the gap is:

$$h_{sur} = \frac{2}{r_o} \left( \frac{k_g k_{UZH} k_{rev}}{k_{UZH} k_{rev} - k_g k_{rev} - 2 k_g k_{UZH} \ln(r_2/r_1)} \right)$$

The graph of heat transfer coefficient through the gap is shown in Fig. 7, as a function of the reactor power. This figure also shows three theoretical values recommended by General Atomic for the heat transfer coefficient [2].
FUEL ROD TEMPERATURE PROFILE

From the temperature in the center of the fuel and using the equations of conduction for the fuel element geometry, it is possible to obtain the radial temperature distribution in the fuel element. Figure 8 shows the experimental radial profile of maximum fuel temperature in position B1 and it is compared with the PANTERA code results [14]. The instrumented fuel element was used to measure the fuel temperature at several reactor powers, the results are shown in Fig. 9.
Amir Zacarias Mesquit

FIG. 8. Experimental fuel rod radial temperature profile in position B1 at 265 kW

FIG. 9. Experimental fuel rod radial temperature profile in position B1 at several reactor powers

CRITICAL HEAT FLUX AND DNB

In the fully developed nucleate boiling regime, it is possible to increase the heat flux without, an appreciable change in the surface temperature until the point of DNB. At this point, the bubble motion on the surface becomes so violent that a hydrodynamic crisis occurs with the formation of a continuous vapour film in the surface and the critical heat flux (CHF) is reached. In subcooled boiling the CHF is a function of the coolant velocity, the degree of subcooling, and the pressure. There are a lot of correlations to predict the CHF. The used equation is given by Bernath [15] [16]. This correlation predicts CHF in the subcooled boiling region and is based on the critical wall superheat condition at burnout and turbulent mixing convective heat transfer. Bernath’s equation gives the minimum results so it is the most conservative. It is given by:

\[ q_{cr}^* = h_{cr} (T_{cr} - T_f) \]  \hspace{1cm} (9)

where,
Amir Zacarias Mesquita

\[ h_{cr} = 61.84 \frac{D_w}{D_s + D_i} + 0.01863 \frac{23.53}{D_w} u \]  

and,

\[ T_{cr} = 57 \ln \left( \frac{p - 54}{p + 0.1034} \right) + 283.7 - \frac{u}{1.219} \]  

\( q_{cr} \) is the critical heat flux [W/m²], \( h_{cr} \) is the critical coefficient of heat transfer [W/m²K], \( T_{cr} \) is the critical surface temperature [°C], \( T_f \) is the bulk fluid temperature [°C], \( p \) is the pressure [MPa], \( u \) is the fluid velocity [m/s] \( (u = \frac{\text{channel area}}{\text{water density}}) \), \( D_s \) is the wet hydraulic diameter [m], \( D_i \) is the diameter of heat source [m]. This correlation is valid for circular, rectangular and annular channels, pressure of 0.1 to 20.6 MPa, velocity between 1 to 16 m/s and hydraulic diameter of 0.36 to 1.7 cm.

In reactor power of 265 kW operating in steady state, the core inlet temperature was 47 °C. The critical flow for the Channel 0 is about 1.6 MW/m², giving a DNBR of 8.5. Figure 10 shows the values of critical flow and DNBR for the two channels. The theoretical values for reactor TRIGA of the University of New York [2] and calculated with the PANTERA code [14] for the IPR-R1 are also shown. The two theoretical calculations gave smaller results than the experiments. These differences are due to the temperature used in the models.

![Critical Heat Flux and DNBR as a function of the inlet coolant temperature](image)

**FIG. 10.** Critical heat flux and DNBR as a function of the inlet coolant temperature

**CONCLUSION**

Pool temperature depends on reactor power, as well as on the external temperature because it affects the heat dissipation rate in the cooling tower. Subcooled pool boiling occurs above approximately 60 kW on the cladding surface in the central channels of the IPR-R1 TRIGA core. However, the high heat transfer coefficient due to subcooled boiling causes the cladding temperature be quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. The IPR-R1 TRIGA Reactor normally operates in the range from 100 kW until a maximum of 250 kW. On these power levels the heat transfer regime between the clad surface and the coolant is subcooled nucleate boiling in the hottest fuel element. Boiling heat transfer is usually the most efficient heat transfer pattern in nuclear reactors core [6]. The results can be considered as typical of pool-type research reactor.

The minimum DNBR for IPR-R1 TRIGA (DNBR=8.5) is much larger than other TRIGA reactors. The 2MW McClellen TRIGA [17] has a DNBR=2.5 and the 3 MW Bangladesh TRIGA has a DNBR=2.8 [18]. The power reactors are projected for a minimum DNBR of 1.3. In routine operation they operated with a DNBR close to 2. The IPR-R1 reactor operates with a great margin of safety at its present power of 250 kW, the maximum heat flux in the hottest fuel is about 8 times lesser than the
critical heat flux that would take the hydrodynamic crisis in the fuel cladding. This investigation indicates that the reactor would have an appropriate heat transfer if the reactor operated at a power of about 1 MW. The data showed the efficiency of the natural circulation to remove the heat generated by the fissions in the core.

ACKNOWLEDGEMENTS

The author would like to express their special thanks to the IPR-R1 TRIGA operator team for their help during the experiments and the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the financial support.

REFERENCES

EXPERIMENTAL INVESTIGATION OF THE ONSET OF NUCLEATE BOILING IN THE IPR-R1 TRIGA NUCLEAR RESEARCH REACTOR

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ABSTRACT
The IPR-R1 TRIGA Nuclear Research Reactor at the Nuclear Technology Development Center (CDTN), in Belo Horizonte (Brazil), is a pool type reactor cooled by natural circulation. Fuel to coolant heat transfer patterns must be evaluated as function of the reactor power in order to evaluate the thermal hydraulic performance of the core. The heat generated by nuclear fission in the reactor core is transferred from fuel elements to the cooling system through the fuel/cladding (gap) and the cladding to coolant interfaces. As the reactor core power increases the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to subcooled nucleate boiling. Experimental results indicated that subcooled pool boiling occurs at the cladding surface in the reactor core central channels at power levels in excess of 60 kW. However, due to the high heat transfer coefficient in subcooled boiling the cladding temperature is quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. An operational computer program and a data acquisition and signal processing system were developed as part of this research project to allow on line monitoring of the operational parameters.

INTRODUCTION
The IPR-R1 TRIGA (Training, Research, Isotopes, General Atomic), showed in Fig. 1 is a pool type nuclear research reactor, with an open water surface and the core has a cylindrical configuration. The maximum core power is 250 kWth, cooled by light water and with graphite reflectors. The fuel is an alloy of zirconium hydride and uranium enriched at 20% in $^{235}$U. The reactor core, showed in Fig. 2, has 63 cylindrical fuel elements, 58 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements with 20 % enrichment and 8.5 wt % uranium. One of these steel-clad fuel elements is instrumented with three thermocouples along its center axis (Fig. 3).

Experimental studies have been performed in the IPR-R1 Reactor [1] to find out the core thermal power, the temperature distribution as a function of the reactor power under steady-state conditions, the flow distribution in the coolant channels, the heat transfer coefficient on the heated surface and a prediction of critical heat flux.

The objective of the thermal and hydrodynamic projects of the reactors is to remove the heat safely, without producing excessive temperature in the fuel elements. The regions of the reactor core where boiling occurs at many different power levels can be determined from the heat transfer coefficient data.

The thermal conductivity ($k$) of metallic alloys is mainly a function of temperature. In nuclear fuels, this relationship is more complicated because $k$ also becomes a function of irradiation as a result of the changes in the chemical and physical composition (porosity changes due to temperature and fission products). Many factors affect the fuel thermal conductivity. The major factors are temperature, porosity, oxygen to metal atom ratio, PuO$_2$ content, pellet cracking, and burnup. The second largest resistance to heat conduction in the fuel rod is due to the gap. Several correlations exist to evaluate its value in power reactors fuels, which use mainly uranium oxide [2]. The only reference found to TRIGA reactors fuel is General Atomic [3] that recommends the use of three hypotheses for the heat transfer coefficient through the gap. The heat transfer coefficient ($h$) is a property not only of the system but it also depends on the fluid properties. The determination of $h$ is a complex process that depends on the thermal conductivity, density, viscosity, velocity, dimensions and specific heat. All these parameters are temperature-dependent and change when heat is being transferred from the

Figure 1 – Upper view of the IPR-R1 TRIGA Reactor
heated wall to the fluid. An operational computer program and a data acquisition and signal processing system were developed as part of this research project to allow on line monitoring of the operational parameters [4].

As the reactor core power increases, the heat transfer regime from the fuel cladding to the coolant changes from single-phase natural convection to subcooled nucleate boiling.

![Image](image.png)

**Figure 2 - Core top view with the instrumented fuel element in ring B**

**OVERALL THERMAL CONDUCTIVITY OF THE FUEL ELEMENTS**

From Fourier equation described in [5, 6], it was obtained the expression for overall thermal conductivity \(k_o\), in [W/mK], for cylindrical fuel elements

\[
k_o = \frac{q'' r^2}{4(T_o - T_{sur})},
\]

(1)

where \(q''\) is the volumetric rate of heat generation [W/m³], \(T_o\) and \(T_{sur}\) are the fuel center temperature and the surface temperature [°C] and \(r\) is the fuel element radius [m].

The temperature at the center of the fuel was measured. The heat transfer regime at the power of 265 kW in all fuel elements is the subcooled nucleate boiling. The cladding outside temperature is the water saturation temperature (\(T_{sat}\)) at the pressure of 1.5 bar (atmospheric pressure added up of the water column of ~ 5.2 m), increased of the wall superheat (\(\Delta T_{sat}\)). The superficial temperature (\(T_{sur}\)) in [°C] is found using the expression below, where \(T_{sat}\) is equal to 111.37 °C [7].

\[
T_{sur} = T_{sat} + \Delta T_{sat}.
\]

(2)

The wall superheat is obtained by using the correlation proposed by McAdams found in [8],

\[
\Delta T_{sat} = 0.81q''^{0.259},
\]

(3)

with \(q''\) in [W/m³] and \(T_{sat}\) in [°C].

A fuel element instrumented with three type K thermocouples was introduced into position B1 of the core. Figure 3 show the diagram of the instrumented fuel element and the Table 1 presents its mean characteristics. Figure 4 shows the core configuration, two thermocouples were placed in two core channels adjacent to ring B.

![Diagram](image.png)

**Figure 3 - The instrumented fuel element**

<table>
<thead>
<tr>
<th>Table 1 – Instrumented fuel element features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter</td>
</tr>
<tr>
<td>-------------------------------------------</td>
</tr>
<tr>
<td>Heated length</td>
</tr>
<tr>
<td>Outside diameter</td>
</tr>
<tr>
<td>Active outside area</td>
</tr>
<tr>
<td>Fuel outside area (U-ZrH₁₋₋)</td>
</tr>
<tr>
<td>Fuel element active volume</td>
</tr>
<tr>
<td>Fuel volume(U-ZrH₁₋₋)</td>
</tr>
<tr>
<td>Power (total of the core = 265 kW)</td>
</tr>
</tbody>
</table>
The mass flow rate is given indirectly from the thermal balance along the channel using measurements of the water inlet and outlet temperatures:

\[ q = \dot{m} c_p \Delta T, \]  

where \( q \) is the power supplied to the channel [kW]; \( \dot{m} \) is the mass flow rate in the channel [kg/s]; \( c_p \) is the isobaric specific heat of the water [J/kgK]; and, \( \Delta T \) is the temperature difference along the channel [°C].

### Table 2. Channel 0 and Channel 1' characteristics [1]

<table>
<thead>
<tr>
<th></th>
<th>Channel 0</th>
<th>Channel 1'</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Area (A)</td>
<td>1.574</td>
<td>8.214</td>
<td>cm²</td>
</tr>
<tr>
<td>Wetted Perimeter (Pw)</td>
<td>5.901</td>
<td>17.643</td>
<td>cm</td>
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<td>1.862</td>
<td>cm</td>
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</tr>
<tr>
<td>B6 and C12 Fuel Diameter (Al)</td>
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<td>3.73</td>
<td>cm</td>
</tr>
<tr>
<td>C1 Control Rod Diameter</td>
<td>3.80</td>
<td>3.80</td>
<td>cm</td>
</tr>
<tr>
<td>Central Thimble</td>
<td>3.81</td>
<td>3.81</td>
<td>cm</td>
</tr>
<tr>
<td>Core Total Power (265kW)</td>
<td>100</td>
<td>100</td>
<td>%</td>
</tr>
<tr>
<td>B1 Fuel Contribution</td>
<td>0.54</td>
<td>1.11</td>
<td>%</td>
</tr>
<tr>
<td>B6 Fuel Contribution</td>
<td>0.46</td>
<td>0.94</td>
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<td>C12 Fuel Contribution</td>
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<tr>
<td>Channel Total Power</td>
<td>1.00</td>
<td>3.70</td>
<td>%</td>
</tr>
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</table>

The reactor was operated on steps of about 50 kW until 265 kW and data were collected in function of the power supplied to Channel 1' and Channel 0. The values of the water thermodynamic properties at the pressure 1.5 bar as function of the bulk water temperature at the channel were taken from Wagner and Kruse [7]. The curve for heat transfer coefficient \( h_{sw} \) in the single-phase region is shown in Fig. 7 as function of the power and Table 3 presents the water properties.
### Table 3 – Cooling properties and single phase convective coefficient

<table>
<thead>
<tr>
<th>(q) Core [kW]</th>
<th>(q) Channel [kW]</th>
<th>(\Delta T) [°C]</th>
<th>(c_p) [kJ/kgK]</th>
<th>(vi) [kg/s]</th>
<th>(G) [kg/m³s]</th>
<th>(u) [m/s]</th>
<th>(\mu) [10⁻³kg/ms]</th>
<th>(k) [W/mK]</th>
<th>Re</th>
<th>Pr</th>
<th>(h_{sur}) [kW/m²K]</th>
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<tbody>
<tr>
<td>265</td>
<td>9.81</td>
<td>13.9</td>
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<td>6081</td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>(q) Core [kW]</th>
<th>(q) Channel [kW]</th>
<th>(\Delta T) [°C]</th>
<th>(c_p) [kJ/kgK]</th>
<th>(vi) [kg/s]</th>
<th>(G) [kg/m³s]</th>
<th>(u) [m/s]</th>
<th>(\mu) [10⁻³kg/ms]</th>
<th>(k) [W/mK]</th>
<th>Re</th>
<th>Pr</th>
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### Subcooled Nucleate Boiling Region

For the subcooled nucleated boiling region (local or surface boiling), the expression used is shown below, according to [10, 11]:

\[
h_{sur} = \frac{q''}{\Delta T_{sat}}
\]

(6)

where \(h_{sur}\) is the convective heat-transfer coefficient from the fuel cladding outer surface to the water [kW/m²K]; \(q''\) is the fuel surface heat flux [kW/m²]; and, \(\Delta T_{sat}\) is the surface superheat in contact with the water [°C].

Table 4 presents the thermal parameter of the fuel element. Figure 6 shows the fuel element surface heat transfer coefficient for the coolant as a function of the superheat, in both regimes. This curve is specific for the IPR-R1 TRIGA reactor conditions. The transition point between single-phase convection regime to subcooled nucleate boiling regime (onset of nucleate boiling - ONB) is approximately 60 kW as shown in the graph.

### Table 4 – Fuel element thermal parameter

<table>
<thead>
<tr>
<th>(q) Core [kW]</th>
<th>(q) Eff [W]</th>
<th>(T_e) [°C]</th>
<th>(q') [W/m]</th>
<th>(q'') [W/m²]</th>
<th>(\Delta T_{sat}) [°C]</th>
<th>(T_{sur}) [°C]</th>
<th>(k_e) [W/mK]</th>
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<td>18391</td>
<td>155690</td>
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<td>8.44</td>
<td>15.0</td>
<td>8.31</td>
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</table>
Figure 7 presents the curves for the heat transfer coefficient \( (h_{\text{fuel}}) \) on the fuel element surface and for the overall thermal conductivity \( (k) \) in fuel element as function of the power, obtained with the instrumented fuel at core ring B.

![Graph showing heat transfer coefficient and thermal conductivity](image)

**CONCLUSION**

The IPR-R1 TRIGA Reactor normally operates in the range from 100 kW until a maximum of 250 kW. On these power levels the heat transfer regime between the clad surface and the coolant is subcooled nucleate boiling in the hottest fuel element. The transition point between single-phase convection regime to subcooled nucleate boiling regime (onset of nucleate boiling - ONB) is approximately 60 kW on the cladding surface in the central channels of the IPR-R1 TRIGA core. However, the high heat transfer coefficient due to subcooled boiling causes the cladding temperature to be quite uniform along most of the active fuel rod region and do not increase very much with the reactor power. Boiling heat transfer is usually the most efficient heat transfer pattern in nuclear reactors core. The results can be considered as typical of pool-type research reactor.

Pool temperature depends on reactor power as well as on the external temperature since it affects the heat dissipation rate in the cooling tower. The data showed the efficiency of the natural circulation to remove the heat generated by the fissions in the core.

**ACKNOWLEDGEMENTS**

The author would like to express their special thanks to the IPR-R1 TRIGA operator team for their help during the experiments and the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the financial support.

**NOMENCLATURE**

<table>
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<tr>
<th>Symbol</th>
<th>Quantity</th>
<th>SI Unit</th>
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<tr>
<td>( A )</td>
<td>Area</td>
<td>( \text{m}^2 )</td>
</tr>
<tr>
<td>( c_p )</td>
<td>Isobaric specific heat</td>
<td>( \text{J/kgK} )</td>
</tr>
<tr>
<td>( D_w )</td>
<td>Hydraulic diameter</td>
<td>( \text{m} )</td>
</tr>
<tr>
<td>( D_s )</td>
<td>Hydraulic Diameter</td>
<td>( \text{cm} )</td>
</tr>
<tr>
<td>( G )</td>
<td>Mass flow</td>
<td>( \text{kg/m}^2\text{s} )</td>
</tr>
<tr>
<td>( h )</td>
<td>Convective heat-transfer coefficient</td>
<td>( \text{kW/m}^2\text{K} )</td>
</tr>
<tr>
<td>( k )</td>
<td>Thermal conductivity</td>
<td>( \text{W/mK} )</td>
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<td>Mass flow rate</td>
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<td>( P_h )</td>
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<tr>
<td>( q )</td>
<td>Power</td>
<td>( \text{kW} )</td>
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<tr>
<td>( q'' )</td>
<td>Surface heat flux</td>
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<tr>
<td>( q''' )</td>
<td>Volumetric rate of heat</td>
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<tr>
<td>( T )</td>
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<tr>
<td>( \mu )</td>
<td>Fluid dynamic viscosity</td>
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**REFERÊNCIAS**


Experimentos Neutrônicos e Termohidráulicos no TRIGA IPR-R1

Amir Zacarias Mesquita (amir@cdtn.br)

RESUMO

O objetivo do projeto é a realização de experimentos neutrônicos e termohidráulicos para conhecimento dos parâmetros operacionais do reator TRIGA IPR-R1. Como subsídio aos experimentos é importante a atualização da instrumentação de operação do reator e implantação de novos equipamentos para medida e controle das suas variáveis de operação, destacando-se a necessidade da presença de elementos combustíveis instrumentados com sensores de temperatura posicionados no núcleo, para monitoração de sua temperatura máxima. A realização dos experimentos visa também aumentar a segurança e a confiabilidade na operação do reator, na nova potência de 250 kW, mantendo também as competências técnicas teórica e experimental do CDTN em neutrônica e termohidráulica.

Palavras-chave: neutrônica, termohidráulica, reator nuclear de pesquisa, TRIGA, reatividade.

INTRODUÇÃO

A realização de pesquisas em neutrônica e termohidráulica no IPR-R1 visa, principalmente, o conhecimento de seus parâmetros operacionais. A implementação do projeto atende às recomendações da AIEA quanto a modernização do controle dos reatores de pesquisa para monitoramento dos seus limites operacionais, contempla também as diretrizes das normas ISO relativas à confiabilidade dos laboratórios [1] [2]. Foram realizados diversos experimentos neutrônicos e termohidráulicos, tais como a calibração das barras de controle e da potência térmica liberada, os quais, conforme documentação do IPR-R1, devem ser realizados anualmente. Concluiu-se uma dissertação sobre o mapeamento do fluxo neutrônico nos terminais de irradiiação. Um trabalho inédito realizado foi o mapeamento da reatividade negativa introduzida por um vazio simulado inserido entre os elementos combustíveis. Esta informação é importante devido ao surgimento de ebulição nucleada sub-restriada nos canais do núcleo na operação em 250 kW. Novas metodologias também foram testadas, como o método calorimétrico para o cálculo da potência térmica. Como o iPR-R1 ainda não esta licenciado para operar em 250kW, os experimentos foram realizados no máximo em 100kW. Os resultados não serão apresentados, pois foram realizados em fev/mar 2009, estando em fase de análise. Relacionado indiretamente ao Projeto, houve a contribuição de dois pesquisadores, na elaboração do Relatório final do Grupo de
Engenharia de Reatores do Reator Multipropósito Brasileiro - RBM.

A seguir são descritas as realizações de cada Tarefa do Projeto no último ano.

DETERMINAÇÃO DE PARÂMETROS NEUTRÔNICOS

Realizaram-se os seguintes experimentos no reator [3]: calibração da barra de Regulação e intercalibração das barras de Controle e Segurança; medidas do excesso de reatividade e da margem de desligamento; medidas dos coeficientes de reatividade de temperatura, de potência, e de vazio, em várias posições radiais do núcleo; medida do déficit de potência, e levantamento do envenenamento por xenônio.

Relacionado à tarefa, apresentou-se dois trabalhos em eventos internacionais [4] [5] e um artigo encontra-se em fase de publicação, em revista internacional [6]. A responsável pela tarefa recebeu convite do Ministério de Minas e Energia da Colômbia e da AIEA para participar, como perito, da avaliação final para concessão das licenças dos operadores do reator TRIGA IAN-R1 aquele país [7]. Por falta de tempo hábil para implementar a documentação de afastamento do país, o convite não pode ser aceito.

DETERMINAÇÃO DE PARÂMETROS TÉRMICAMETRÍCOS

Após a realização dos testes descritos no item anterior, realizaram-se os seguintes experimentos termohidráulicos [3]: avaliação da potência pelos métodos calorimétrico, balanço térmico no primário; balanço térmico no secundário e avaliação do fluxo e da velocidade do refrigente no canal quente do núcleo. Estavam previstos, mas não foram realizados, os seguintes ensaios: avaliação da potência pelo monitoramento da temperatura do centro do combustível [8] (ainda não foram recuperados todos os termopares do combustível instrumentado) e o monitoramento da condutividade elétrica e do pH da água do circuito primário.

Relacionado à tarefa, apresentaram-se dois trabalhos em eventos nacionais [9] [10] e um trabalho em evento internacional [11]. Um artigo foi submetido a uma revista internacional e encontra-se em fase de análise [12]. Um projeto de pesquisa foi submetido ao CNPq, não sendo aprovado [13].

MAPEAMENTO DE FLUXO DE NÉUTRONES NOS TERMINAIS DE IRRadiação

Esta tarefa contempla uma dissertação de mestrado no Curso de Eng. Nuclear da UFMG. O objetivo é o levantamento do fluxo de nêutrons nos locais de irradiação de
amostras do IPR-R1. O trabalho foi concluído com defesa no dia 30 março 2009 [14]. Uma publicação em revista internacional encontra-se em fase de publicação [15].

ESTUDO DE INSTRUMENTAÇÃO E DE CONTROLE DIGITAL DE REATORES

Esta tarefa objetiva atualizar a instrumentação de medida e controle do reator e manter o banco de dados eletrônico de monitoramento de suas variáveis operacionais [18]. Integra também a tarefa o estudo e o desenvolvimento de técnicas digitais, tendo como meta a troca do controle do IPR-R1 de analógico para digital. A implantação de lógica microprocessada seguirá a tendência mundial, sendo recomendação da ANEP [1]. Solicitou-se à General Atomic, fabricante do TRIGA, a especificação de um sistema de controle para o IPR-1 [16]. O sistema oferecido utiliza processadores digitais, monitores de LCD e com um canal de potência originado do combustível instrumentado. O preço do sistema é de cerca de US$ 1 300 000.

Com relação a esta tarefa, foi aprovado, junto à FAPEMIG um projeto de pesquisa (Programa Pesquisador Mineiro - PPM II) [17]. Entre os equipamentos já adquiridos citam-se: dois indicadores digitais de condutividade e de temperatura (com sondas), medidor de pH, relógio e cronômetros digitais, placas e software de aquisição de dados, computadores e monitores de vídeo LCD 22". Apresentou-se um trabalho em evento internacional [18]. Redigiram-se relatórios técnicos com propostas de atualização do sistema de controle do IPR-R1 [19] [20]. Uma orientação de iniciação científica, com bolsa da FAPEMIG, foi encerrada no período [21]. Encontra-se em andamento dissertação de mestrado no Curso de Pós-graduação do CDTN [22].

RECUPERAÇÃO E COMPRA DE ELEMENTO COMBUSTÍVEL INSTRUMENTADO

O objetivo desta tarefa é recuperar os termopares do elemento combustível instrumentado e adquirir um novo elemento, para monitorar a temperatura máxima do núcleo. Obteve-se sucesso parcial na recuperação dos termopares [8]. Solicitou-se a General Atomic orçamento para a compra de elemento combustível padrão e instrumentado [23]. Apresentou-se projeto de pesquisa ao CNPq para aquisição de combustível instrumentado (87000 €), não sendo aprovado [24].

CONCLUSÃO

Os reatores TRIGA (Training, Research, isotopes, General Atomic), devido às suas características de segurança, são os reatores mais apropriados para a realização
de pesquisas em neutrônica, termohidráulica, instrumentação e treinamento. O TRIGA IPR-R1 queimou, em 49 anos de operação, apenas cerca de 4% de seu combustível. Portanto, é um reator com grande potencial de uso. A potência inicial do IPR-R1 era de 30 kW, acrescentaram-se elementos combustíveis subindo a potência para 100 kW. Em 2002, acrescentaram-se novos combustíveis, atingindo-se níveis de 250 kW. Em 2007 e 2008, enviou-se o Relatório de Análise de Segurança e o Manual de Operação para a CNEN, com todas as não conformidades atendidas e aguarda-se o licenciamento para a operação na nova potência [25] [26]. A operação do reator a 250 kW possibilitará a pesquisa de novos parâmetros, conforme exemplo de outros reatores TRIGA, sendo, para tanto, necessário a atualização da instrumentação de monitoramento das variáveis, como por exemplo, a utilização de elementos combustíveis instrumentados.

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2. Responsável pela Tarefa: Testes Neutônicos.
IPR-R1 TRIGA RESEARCH REACTOR DECOMMISSIONING: PRELIMINARY PLAN

Clédola Căssia Oliveira de Tello¹, Pablo Andrade Grossi² and Amir Zacarias Mesquita³
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ABSTRACT

The International Atomic Energy Agency (IAEA) is concerning to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes. In this way the IAEA recommends that decommissioning planning should be part of all radioactive installation licensing process. There are over 200 research reactors that have either not operated for a considerable period of time and may never return to operation or, are close to permanent shutdown. Many countries do not have a decommissioning policy, and like Brazil not all installations have their decommissioning plan as part of the licensing documentation. The Nuclear Technology Development Centre (CDTN/CNEN) has a TRIGA Mark I Research Reactor in operation for 47 years with 3.6% average fuel burn-up. The original power was 100 kW and it is being licensed for 250 kW, and as no decommissioning policy was adopted, it needs to do the decommissioning plan for it now. This paper presents the description of IPR-R1 TRIGA reactor and the preliminary plan for its decommissioning, as part of the licensing requirements.

1. INTRODUCTION

All nuclear installations should be commissioned, and after their shutdown they should be decommissioned. There are many factors that bring to the decision to shutdown a nuclear installation, for example obsolescence, security, regulatory aspects, political changes, accidents, low performance, etc. There are over 200 research reactors that have either not operated for a considerable period of time and may never return to operation or, are close to permanent shutdown [1, 2].

Decommissioning is defined as all administrative and technical actions that should be taken at the end of a nuclear installation in order to assure the suitable physical and radiological protection to the workers, general public, and environment [3]. These actions allow also the removal of the installation from the regulatory control. This process involves two phases: decontamination and dismantling. The decontamination is the phase in which the complete or partial removal of contamination is done by a physical, chemical or biological process. The dismantling consists on the disassembly and removal of any structure, system or component during decommissioning. Dismantling may be performed immediately after permanent retirement of a nuclear facility or it may be postponed.

The decommissioning plan is the document, in which is organized all information about the proposed decommissioning activities for the facility. It allows the regulatory body to make a proper evaluation and to ensure that decommissioning of the facility can be performed in a safe manner. IAEA is concerned to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful
purposes. In this way the IAEA recommends that decommissioning planning should be part of all radioactive installation licensing process. Many countries do not have a decommissioning policy, and like Brazil not all installations have their decommissioning plan as part of the licensing documentation.

Currently the search of the sustainable development proposes that the potential for redevelopment should not be ignored. Sustainable development implies the need to combine economic development with conservation of natural resources such as land. In the case of decommissioning, the recycling of land implied by redevelopment of a site offers a valuable means of avoiding the need to obtain further "greenfield" sites. This also implies economic development with the maintenance of social and community integrity. Both of these benefits can be attained by the sensitive and organized redevelopment of sites to provide continuity of employment and new production opportunities. Finally, the principles of sustainable development suggest a more transparent and participative decision making process than has been the practice in many aspects of nuclear development [4].

The Nuclear Technology Development Centre (CDTN/CNEN) has a TRIGA Mark I Research Reactor in operation for 47 years with 3.6% average fuel burn-up. The original power was 100 kW and it is being licensed for 250 kW, and as no decommissioning policy was adopted, it needs to do the decommissioning plan for it. This paper presents the description of IPR-R1 TRIGA Reactor and the preliminary plan for its decommissioning, as part of the licensing requirements.

2. IPR - R1 TRIGA REACTOR

The first and very important step to establish the decommissioning plan is to know and to define the installation to be decommissioned. The 250 kW TRIGA reactor (IPR-R1) is at the Nuclear Technology Development Centre (Centro de Desenvolvimento de Tecnologia Nuclear – CDTN) on the campus of Federal University of Minas Gerais, in Belo Horizonte, and is mainly used for research purposes. The first criticality was achieved on November of 1960. The regime of operation of the reactor is about 4 hours per day, 4 days per week, and 40 weeks per year. The integrated burn-up of the reactor since its first criticality is about 130 MW-Days. In the Fig. 1 is presented external and internal aspects of the TRIGA IPR-R1 reactor.

![Figure 1. Top View of the TRIGA IPR-R1 Core](image-url)
IPR-R1 Reactor TRIGA is installed in a building especially constructed to shelter it in a reinforced concrete structure. The main characteristics of the fuel element and of the reactor are presented in Table 1. The IPR-R1 has no short and medium range storage problems, due to its low nominal power. The first fuel assembly replacement is expected to occur only in 2010. In Table 2 is presented the IPR-R1 spent fuel assembly (SFA) inventory.

A policy regarding spent fuel or high-level waste disposal was not yet defined by Brazilian government. However, given that the legal framework regarding waste disposal is being defined, this issue will be discussed at the national level.

<table>
<thead>
<tr>
<th>Table 1. Fuel element and reactor characteristics [5]</th>
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</thead>
<tbody>
<tr>
<td>Fuel Element Type</td>
</tr>
<tr>
<td>Number</td>
</tr>
<tr>
<td>Geometry</td>
</tr>
<tr>
<td>Active length</td>
</tr>
<tr>
<td>Cladding material</td>
</tr>
<tr>
<td>Cladding thickness</td>
</tr>
<tr>
<td>Cladding diameter</td>
</tr>
<tr>
<td>Fuel diameter (U-ZrH)</td>
</tr>
<tr>
<td>Fuel-moderator material</td>
</tr>
<tr>
<td>Amount of U (% weight)</td>
</tr>
<tr>
<td>Enrichment (% 235U)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Table 2. SFA inventory of the IPR-R1 Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Facility</td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td>IPR-R1</td>
</tr>
</tbody>
</table>

FE = fuel element  RR = research reactor

3. DECOMMISSIONING PLAN

A plan shall be prepared to ensure safety and feasibility throughout the decommissioning. This plan shall be submitted to the safety committee and regulatory body for approval by before beginning of decommissioning activities. The decommissioning plan shall include an evaluation of one or more decommissioning alternatives suitable for the studied installation and that achieve the requirements of the regulatory body [1, 2, 4, 6]. Examples of alternative approaches to decommissioning are:
a) Protective storage in an intact condition after removal of all fuel assemblies and readily removable radioactive components and wastes,

b) Entombment of radioactive structures and large components after removal of all fuel assemblies and readily removable radioactive components and wastes; and

c) Removal of all radioactive materials and thorough decontamination of the remaining structures to permit unrestricted use.

3.1. General Aspects of a Decommissioning Plan

Aspects of the facility design shall be reviewed while developing the plan, mainly these ones that can optimize the decommissioning. It is also important to evaluate its operational history, including the changes in the geometry of the core, accidents, changes in the design, registry of the maintenance and operation of systems and equipment, registry of the development and techniques used during the shutdown for maintenance, occupational doses, experience of the personnel (documented), generated wastes evaluation etc.

The decommissioning plan shall include all steps that lead to eventual complete decommissioning to the point that safety can be ensured with minimum or no surveillance. These stages may include storage and surveillance, restricted site use and unrestricted site use. As the decision for the decommissioning often is made after a period of extended shutdown, occurrences during this period shall be considered when developing the decommissioning plan. It should be used the experience of the operation personnel. The waste estimation (volume, weight, activity, classification), which will be generated during the decommissioning, must be very well known before the establishment of dismantling plan is, because it is larger and different than that generated during the reactor operation and this must be in accordance with the repository design. Concerning to the waste no matter what is the decommissioning strategy, a factor that has a big influence in both economical and environmental aspects is to minimize (e.g. recycling and reusing) as much as possible the waste generation [1, 2, 6].

Procedures for handling, dismantling and disposal of experimental devices or other irradiated equipment that requires storage and eventual disposal shall be established in advance or if already constructed as early as possible. The purpose of the new technology and different approaches for the decommissioning is to reduce the occupational doses and generate the lesser environmental impact during the transference and removal of the materials and components. All activities during the decommissioning process shall be subjected to a QA program.

3.2. Documentation for Decommissioning Operation Licensing

Each licensee is responsible for ensuring that relevant national rules and regulations are applied to the circumstances of the facility being decommissioned. Nevertheless, the following contents shall be take part of the decommissioning documentation [1, 2, 6]:

✓ Introduction;

✓ Facility description: Physical description of the site and facility, operational history, systems and equipment, radioactive and toxic material inventory;

NAC 2007, Santa, SP, Brazil.
✓ Decommissioning strategy: Objectives; decommissioning alternatives; safety principles and criteria; waste type, volumes and routes; dose estimates; cost estimates; financial arrangements; selection and justification of preferred option;

✓ Project management: resources; organization and responsibilities; review and monitoring arrangements; training and qualifications; reporting and records;

✓ Decommissioning activities: description and schedule of phases and tasks, decontamination activities; dismantling, waste management, surveillance and maintenance programs;

✓ Safety assessment: dose predictions for tasks; demonstration of ALARA for tasks; radiation monitoring and protection system; physical security and materials control; management of safety; risk analysis; operating rules and instructions; justification of safety for workers, general population, and environment;

✓ Environmental impact assessment;

✓ Quality assurance program;

✓ Radiation protection and safety program;

✓ Final radiation survey proposal;

✓ Outline of the final decommissioning report: Summary of work; demonstration of compliance with requirements;

✓ Continued surveillance and maintenance and future decommissioning activities (deferred stages of decommissioning).

3.3 IPR-R1 Decommissioning

The guidelines for IPR-R1 decommissioning are:

a) It will be followed the safety and environmental principles defined by they regulatory bodies;

b) The preliminary qualitative inventory of radiological and toxic material consists on concrete, aluminum and stainless steel tubes, bombs, fuel elements cladding of Al and stainless steel, U, fission products, Zr, graphite, boron carbide.

c) The procedures for radioactive waste management of CDTN and IAEA will be used. For the hazardous and regular wastes it will be used the procedures established by the environmental institution.

d) As options for the decommissioning can be considered: removal of the fuel assemblies and decontamination for following restricted uses or removal of all radioactive materials and thorough decontamination of the remaining structures to permit unrestricted use.

e) Just little time before the shutdown the decommissioning will be selected in accordance with the actual legislation, and the political and economical situation. As the reactor is at CDTN's site probably the strategy will be: take off the fuel elements and the intern of the reactor. The concrete will be classified and the area can be used for other applications, because the core is in an excavated hole.

f) The packages will be selected among the qualified ones, in accord of the waste to be conditioned.
g) The equipment and staff requirements will be defined depending on the decontamination activities and on the material defined as radioactive waste.

h) The safety and environmental assessments will be done by CDTN’s radiological and environmental protection staff, respectively.

i) Costs should be estimated in advance, so that it would be provided the necessary budget.

j) A special report with the decommissioning plan should be prepared and sent to the regulatory body just before the reactor shutdown requesting the license for that.

4. CONCLUSIONS

The decommissioning plan should take part of the documentation presented to commission nuclear installations. The IPR-R1 a research reactor operating at CDTN/CNEN is being commissioned for 250 kW. The draft of the decommissioning plan for it is being written.

The decommissioning plan should take part of the documentation presented to commission nuclear installations. In the initial IPR-R1 licensing, the decommissioning aspects were not considered and no decommissioning plan was developed during the commissioning activities. Nowadays, the reactor operating at CDTN/CNEN is being commissioned for operation in 250 kW. The draft of the decommissioning plan for it is being written and will take part of this new licensing documentation.

This documentation regarding to the decommissioning planning can be used as a baseline or guide for other radioactive installation either for licensing future processes or for revision of existent documentation.

5. REFERENCES

Power Calibration of the TRIGA Mark

Introduction

The TRIGA Mark I IPR-RI Nuclear Reactor is a pool type reactor, designed for research, training and radiotrace production. The fuel elements at the reactor core are cooled by water natural convection. The heat removal capability of this process is great enough for safety reasons at the current maximum 250 kW power level of the reactor (Volos, 1999 and Huda et al, 2001). However, a heat removal system is provided for removing heat from the reactor pool water, as shown in Figure 1. The water is pumped through a heat exchanger, where heat is transferred from the primary to the secondary loop. The secondary loop water is cooled in an external cooling tower.

Over the many years since the first TRIGA reactor was built, a number of methodological variations have been evolved for the calibration of the reactor thermal power. The reactor power can be determined from measuring the absolute thermal neutron flux distribution across the core in horizontal and vertical planes. Flux distributions are measured with activation of cadmium covered and bare foils irradiated at steady reactor power (Souza et al., 2002), but it should be noted that this method is time consuming and not accurate (Shaw, 1960). This method is practical only for zero power reactors and in practice it is very seldom performed for other reactors (for the TRIGA IPR-RI Reactor absolute thermal flux distribution was not performed since reactor rebuilding in 1976).

Power monitoring of nuclear reactors is always done by means of neutronic instruments, but its calibration is done by thermal procedures, (Zagar et al., 1999; Jones, Elliott, 1974 and Verri, 1974).

Thermal power calibrations of low power research reactors (up to 1 MW) are normally performed during the initial start-up and their results are used for many years. However, some more fuel rods were added in the core to compensate fuel burning along the years, changing the neutron flux distribution. Moreover, new experimental devices introduced in the reactor pool have changed the overall heat capacity of the system. So it became necessary to calibrate the system annually with the presence of all devices in the core and around it, as accurate power is important for many irradiation experiments.

General Atomic, the TRIGA reactor constructor, traditionally used a methodology for thermal calibration based on the use of a calibrated electrical heater in a calorimetric procedure (Whittomore et al., 1988) at the startup of a number of its facilities. In this methodology the rate of rise of the bulk pit water temperature was measured when using such heaters. The reactor was then operated to give the same rate of rise of water temperature. Thus the reactor power was established at the value produced by the electrical heaters. The experiment should be performed according to the following procedure: minimize hot flow through tank walls during the measurement by selecting the initial and final water temperatures to be about equally above and below the average tank wall temperature; operate the reactor at constant power with primary cooling system switched off; install an appropriate stirrer, record the temperature rise of the pool water; determine the temperature-rise rate ($\Delta T/\Delta t$), and calculate the reactor power as a function of temperature-rise rate.

For most installations, no stirrer was used in these initial power calibrations. Typically a heater with a 10 – 15 kW capacity was used for Mark I and Mark II type reactors. Facilities with larger tanks, such as the 1 MW TRIGA reactor at the Armed Forces Radiological Research Institute (AFMRI) use larger electrical heaters with around 90 – 100 kW capacity (Whittomore et al., 1988). In the calibration of the 250 kW Vienna TRIGA reactor, carried out by Bremsesser et al (1993), it was decided to install five submerged electrical heaters at some selected core positions. These submerged heaters have the same form of the original fuel and the total power was 20 kW. It was found that 19.2 kW of energy was necessary for each 1 °C of water temperature increase.

In most cases, the electrical heater power level was only a tiny fraction of the final reactor power, typically, 10 – 15 kW for 250 – 1000 kW Mark I or Mark II, giving an output which was only 1.2 – 5 % of full power. Even the 90 kW heaters for the 1000 kW AFRRI

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TRIGA reactor gives only about 9% of the full power. Under these circumstances, the extrapolation from calibration power to full reactor power is a factor of at least 10 to 20 or more. Such large scale extrapolations require careful attention to the linearity of the power monitor circuitry, especially in the case of water reflected reactors such as the Mark F TRIGA.

After the first few installations of TRIGA reactors, the initial power calibration for later reactors was performed without the electrical heaters. With the reactor operating at a constant power, the rate of temperature rise was determined. With a tank constant (AT per hour per unit watt) calculated for the applicable heat content of the system, the reactor power was then determined from the measured rate of temperature rise during the reactor operation. Unfortunately, a stirrer was not used in many of these installations resulting in imperfect mixing during these determinations. Without a stirrer and with intermediate reactor power (100-200 kW), the flow pattern of hot water from the core, with primary cooling system switched off, in a columnar chimney rising up about half way to the surface of the pool and then bonding over in a mushroom fashion to return to the region below the reactor core. This was observed by Mosquita et al (2002) during the experiments of the IPR-R1 TRIGA Reactor temperature distribution measurements. At a somewhat higher power level (500 - 700 kW), this columnar chimney of hot water may extend nearly all the way to the top of the reactor tank before turning over to return to the region below the core (Whittmore et al., 1988). Then it is easy to imagine that the measured rate of temperature rise near the top of the pool can give quite different results depending upon where in the tank the temperature probes were located and whether the chimney reaches all the way to the top of the tank. So it is obvious necessary to provide reproducible temperature measurements that are relatively independent of the location of the temperature probe. A stirrer will homogenize the temperature. It is important to note that the stirring produced by the motor driven impeller assures that all the water in the tank participates in the calorimetric measurement. The small rate of energy added by the pump motor is typically less then 1 kW and is negligible for power calibrations performed at 200 - 1000 kW.

The calorimetric procedure is essentially the same whether it involves the calorimetric determination of heat equivalence of electrical energy or the rate of heat generation by a research reactor core. In each case, the calorimeter contains a relatively large volume of water and is constructed with insulated walls to reduce the flow of energy through the calorimeter walls. For the TRIGA system it is obvious, that the heat capacity of the water in the reactor tank is a dominant factor in the whole reactor heat capacity. Zagar et al., (1999) describe that reactor power in research reactors is usually calibrated with an accuracy of 10% and Whittmore et al. (1988) describe that calorimetric power calibration of TRIGA reactors can be obtained with a precision better than ±5%.

However, this method now presents numerous problems for use in the TRIGA Mark I IPR-R1 Nuclear Reactor. The most important difficulty is the removal of the fuel elements and its substitution for electrical heaters that would be quite complex and numerous, due to the great number of facilities already positioned in the reactor pool, above the core. Another difficulty is the presence of all these facilities that makes it difficult to evaluate the bulk thermal capacity of the reactor pool. Third, and almost as important, is the realization that adequate stirring of the water is necessary in order to provide greater precision in the results of the calibration.

The methodology developed for the thermal power calibration consisted of the measurement of the power dissipated at the primary loop and the calculation of the heat losses. The power dissipated at the cooling loop will be closer to the reactor power the closer the water temperature in the reactor pool is to the environment temperature. It means that the reactor pool temperature must be set close to soil temperature around the pool, and that the air temperature in the reactor room set close to the pool temperature (Mosquita and Rezzende, 2001). Therefore, it is important to obtain these conditions and also a stability of the pool temperature over a long period of time, one and a half hours or longer. This can be obtained only after some hours of reactor operation, mainly at night, when there are less changes of the outside air temperature.

The thermal power dissipated in the primary loop can be calculated with a simple thermal balance from the measured values of the inlet and outlet temperatures of the water and its flow rate. We obtain the reactor thermal power by adding this value to the thermal losses. These losses represent a very small fraction of the total power. The power dissipated in the secondary loop was also measured with a thermal balance.

The nuclear power of the TRIGA Mark I IPR-R1 Nuclear Reactor is measured in four different ways:

- The departure channel consists of a fission count with a pulse amplifier that feeds a logarithmic count rate circuit and gives useful power indication from the neutron source level up to a few watts.
- The logarithmic channel consists of a compensated ion chamber feeding a logarithmic (log n) amplifier and recorder and a period amplifier, which gives a logarithmic power indication on a recorder from less than 0.1 W to full power.
- The linear channel consists of a compensated ion chamber feeding a sensitive amplifier and recorder with a range switch, which gives accurate power information from source level to full power on a linear recorder.
- The percent channel consists of an uncompensated chamber feeding a power level monitor circuit and meter, which is calibrated in percentage of full power.

The last three channels were adjusted with the results of the thermal calibration described here.

Nomenclature

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>area of the upper surface of the reactor pool, m²</td>
</tr>
<tr>
<td>C</td>
<td>specific heat capacity, J/kg °C</td>
</tr>
<tr>
<td>C_v</td>
<td>vapor concentration, kg/kg of dry air, dimensionless</td>
</tr>
<tr>
<td>d</td>
<td>thickness of each wall layer, m</td>
</tr>
<tr>
<td>g</td>
<td>acceleration due to gravity, m/s²</td>
</tr>
<tr>
<td>G</td>
<td>Grashof number, dimensionless</td>
</tr>
<tr>
<td>h</td>
<td>depth of the reactor pool, m</td>
</tr>
<tr>
<td>h_c</td>
<td>convective heat transfer coefficient, W/(m²K)</td>
</tr>
<tr>
<td>h_m</td>
<td>mass-transfer coefficient, m³/(m²s)</td>
</tr>
<tr>
<td>k</td>
<td>thermal conductivity, W/(m K)</td>
</tr>
<tr>
<td>L</td>
<td>characteristic length of the heat transfer surface, m</td>
</tr>
<tr>
<td>f</td>
<td>height of the water in the pool</td>
</tr>
<tr>
<td>m</td>
<td>mass flow rate transfer from the pool to the air, kg/s</td>
</tr>
<tr>
<td>n_u</td>
<td>Nusselt number, dimensionless</td>
</tr>
<tr>
<td>P</td>
<td>thermal power, W</td>
</tr>
<tr>
<td>Pr</td>
<td>Prandtl number, dimensionless</td>
</tr>
<tr>
<td>Q</td>
<td>heat losses through the lateral walls, W</td>
</tr>
<tr>
<td>Q_b</td>
<td>heat losses through the bottom, W</td>
</tr>
<tr>
<td>q_wt</td>
<td>heat losses due to the convection, W</td>
</tr>
<tr>
<td>q_v</td>
<td>heat losses due to the evaporation, W</td>
</tr>
<tr>
<td>q_w</td>
<td>flow rate, kg/s</td>
</tr>
<tr>
<td>R</td>
<td>thermal resistance, K/W</td>
</tr>
<tr>
<td>r</td>
<td>radius, m</td>
</tr>
<tr>
<td>S</td>
<td>uncertainty, %</td>
</tr>
<tr>
<td>S_c</td>
<td>Schmidt number, dimensionless</td>
</tr>
<tr>
<td>T</td>
<td>temperature, K</td>
</tr>
</tbody>
</table>

Greek Symbols

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>b</td>
<td>volumetric thermal expansion coefficient of the air, K⁻¹</td>
</tr>
<tr>
<td>ΔT</td>
<td>difference between the temperatures at the inlet and the outlet of the primary loop, °C</td>
</tr>
<tr>
<td>δ</td>
<td>difference between the specific enthalpy of saturated water and the specific enthalpy of saturated steam at the wet bulb temperature of the air in the reactor room, J/kg</td>
</tr>
<tr>
<td>v</td>
<td>kinematic viscosity of the air, m²/s</td>
</tr>
</tbody>
</table>
\[ x = \text{mathematical constant} \ 3.14159, \ \text{dimensionless} \]
\[ \rho_{air} = \text{air density, kg/m}^3 \]

**Subscripts**
- air: relative to air
- al: relative to aluminum layer
- e: relative to convective
- ce: relative to of the external concrete layer
- ci: relative to internal concrete layer
- cool: relative to cooling
- D: relative to the diameter
- s: relative to external radius of wall layer
- ext: relative to external wall of the pool
- i: relative to internal radius of wall layer
- in: relative to internal wall of the pool
- m: relative to coolant water in the primary loop
- out: relative to outlet water with relation of heat exchanger
- p: relative to constant pressure, specific heat of the coolant
- sat: relative to saturation conditions for the air at the reactor room temperature
- ss: relative to stainless steel layer
- sW: relative to water pool surface relative to air at the reactor room
- 1: relative to the lateral walls of the pool
- 2: relative to the bottom of the pool

---

**Thermal Power Dissipated in the Primary Loop**

The thermal power dissipated in the primary loop was obtained through a thermal balance given by the following equation:

\[ Q_{cool} = \dot{m}_{w} \cdot C_{p} \cdot \Delta T \]  

(1)

Where \( \dot{m}_{w} \) is the flow rate of the coolant water in the primary loop, \( C_{p} \) is the specific heat of the coolant, and \( \Delta T \) is the difference between the temperatures at the inlet and the outlet of the primary loop.

The data acquisition system registers the following measurements in each second:
- Temperatures in the pool, in the soil around the pool and in the air at the reactor room;
- Temperatures of the water at the inlet and at the outlet of the primary and secondary loops;
- Flow rate of the coolant water in the primary loop.

The data acquisition computer program calculates the power dissipated in the cooling loop with the collected data being used in Equation 1, and with the \( \dot{m}_{w} \) and \( C_{p} \) values corrected as function of the temperature of the coolant (Miller, 1989).

**Heat Losses from the Reactor Pool to the Environment**

The core of the TRIGA Mark I IPR-R1 Nuclear Reactor is placed below the room floor, in the bottom of a cylindrical pool, 6.625 m deep and 1.92 m in diameter, whose upper surface is 25 cm below the level of the floor. The reactor pool transfers heat to the environment by conduction to the soil, through the horizontal walls and through the bottom of the pool, and by convection and evaporation to the air at the reactor room, through the upper surface.

The reactor pool was built as a five layer cylindrical tank, open at the upper side, as shown in Figure 2. The innermost layer, which is in contact with the water, is 10 mm thick and is made of a special alloy of aluminum (AA-5052-H34). Surrounding it there is a 72 mm thick layer of concrete and then a 6.3 mm thick stainless steel layer. After that, another concrete layer 203 mm thick and finally another stainless steel layer 6.3 mm thick.

**Heat Losses From the Pool to the Soil**

The heat losses through the lateral walls is given by the equation below (Ozisik, 1990)

\[ Q = \frac{T_{in} - T_{out}}{R_{al} + R_{ce} + R_{ci} + R_{m}} \]  

(2)

Where \( T_{in} \) is the average temperature of the internal wall of the pool, \( T_{out} \) is the average temperature of the soil around the reactor, \( R_{al} \) is the thermal resistance of the aluminum layer, \( R_{ce} \) is the thermal resistance of the internal concrete layer, \( R_{ci} \) is the thermal resistance of the stainless steel layer and \( R_{m} \) is the thermal resistance of the external concrete layer.

The thermal resistance for cylindrical walls was obtained from the following equation (Ozisik, 1990)

\[ R = \frac{\ell}{2\pi \cdot k \cdot \ln \left( \frac{r_2}{r_1} \right)} \]  

(3)

Where \( \ell \) is the height of the water in the reactor pool (6.417 m), \( k \) is the thermal conductivity of each material, \( r_1 \) and \( r_2 \) are the internal and external radii of each wall layer.

The heat transfer through the bottom of the pool is obtained from:

\[ Q = \frac{T_{in} - T_{out}}{R_{al} + R_{ce} + R_{ci} + R_{m}} \]  

(4)

The values of the thermal resistance for flat surface section are obtained from the following equation (Ozisik, 1990):

\[ R = \frac{d}{Ak} \]  

(5)


Power Calibration of the TRIGA Mark I Nuclear Research Reactor

Where \( d \) is the thickness of each wall layer and \( A \) is the area of the upper surface.

\[
q_{ev} = m \lambda.
\]

(6)

Where \( \lambda \) is the difference between the specific enthalpy of saturated water and the specific enthalpy of saturated steam at the wet-bulb temperature of the air in the reactor room, and \( m \) is the rate of mass transfer from the pool to the air, given by the equation:

\[
in = h_e \cdot A \cdot \rho_{air} \left( C_{sat} - C_w \right)
\]

(7)

Where \( A \) is the upper surface of the reactor pool, \( \rho_{air} \) is the air density, \( C_{sat} \) is the vapor concentration at saturation conditions for the air at the reactor room temperature, \( C_w \) is the vapor concentration in the air in the reactor room and \( h_e \) is the mass-transfer coefficient given by the following equation:

\[
h_e = \frac{h_c}{\rho_{air} \cdot C_{sat} \cdot Sc^{1/3}}
\]

(8)

Where \( Pr \) is the Prandtl number (0.708 for the air at 25 °C), \( Sc \) is the Schmidt number (0.60 for water vapor diffusing in the air at 25 °C), \( C_{sat} \) is the heat capacity of the air, \( h_c \) is the convection heat transfer coefficient, obtained from:

\[
h_c = \frac{k}{L} = \frac{k}{Nu}
\]

(9)

Where \( k \) is the thermal conductivity in the air, \( L \) is the characteristic length of the heat transfer surface, equivalent to 0.9 times the diameter of the pool or 1.728 m and \( Nu \) is the Nusselt number obtained from:

\[
Nu = 0.14 \left( Gr \cdot Pr \right)^{1/3}
\]

(10)

\( Gr \) is the Grashof number given by:

\[
Gr = \frac{g \cdot \beta \cdot \left( T_{air} - T_w \right)}{\nu^2}
\]

(11)

Where \( g \) is the acceleration due to gravity, \( \beta \) is the volumetric thermal expansion coefficient of the air, \( T_{air} \) is the water pool temperature at the surface, \( T_w \) is the air temperature in the reactor room and \( \nu \) is the kinematic viscosity of the air.

The relative humidity of the air in the room of the reactor was measured, during the tests. The convection heat transfer through the reactor pool surface was calculated with the following equation (Holman, 1963):

\[
q_c = h_c \cdot A \cdot (T_{air} - T_w)
\]

(12)

**Instrumentation**

Two platinum resistance thermometers (PT-100) were positioned at the inlet and at the outlet pipes of the primary cooling loop, just above the water surface of the reactor pool (see \( T_{in} \) and \( T_{out} \) in Figure 1). These thermometers, together with a flow-measuring device at the loop, give the power dissipated through the primary cooling loop. The flow-measuring device consists of an orifice plate and a differential pressure transmitter. This pressure transmitter was calibrated and an adjusted equation was obtained and added to the data acquisition system. The temperature measuring lines were calibrated as a whole, including thermometers, cables, data acquisition cards and computer. The adjusted equations were also added to the data acquisition system.

Two type K thermocouples and one resistance thermometer (PT-100) were positioned inside the pool, at different heights, to measure the water pool temperature. A type K thermocouple was placed just above the pool surface to measure the air temperature at the reactor room. Finally, three type K thermocouples were distributed around the pool, in three holes in the reactor room floor, to measure the soil temperature. The temperature measuring lines were also calibrated as a whole, including thermocouples or resistance thermometer, cables, data acquisition cards and computer. The equations obtained for each line were also added to the data acquisition system.

For the measurement of the power dissipated in the secondary cooling loop, two resistance thermometers (PT-100) were also positioned in its inlet and outlet pipes. The water flow rate at this loop was maintained constant and was also measured.

The sensor signs were sent to an amplifier and multiplexing board, that also makes the temperature compensation for the thermocouples. These signs were sent to a data acquisition card that makes the analog/digital conversion. This card was installed together in a computer where the data were calculated, registered and recorded (Mesquita, 2003). All data were obtained as the average of 120 readings and were recorded together with their standard deviations. The data acquisition system registers these data each second.

**Results**

Initially a thermal power measuring experiment was carried out, with the new reactor core configuration of 63 fuel elements. A thermal power of about 220 kW was measured when the linear channel was indicating the power of 250 kW. So, the ion chambers were replaced and the power indication instruments at the control panel (linear channel, logarithm channel and percent channel) were adjusted again. Another thermal power measuring experiment was then carried out whose results are presented here.
The reactor operated during a period of about 6 hours with a power of 250 kW indicated at the linear channel. The power dissipated through the primary cooling loop was monitored during the whole test period, and the measured temperatures were stable for 1.5 h (from 21:00 h to 22:30 h). Figure 3 shows the evolution of the measured temperatures and Figure 4 shows the evolution of the thermal power dissipated in the primary loop, during the period of stability. Table 1 presents the average thermal power obtained in this same period.

The thermal power dissipated through the primary cooling loop was calculated by Equation (1), as mentioned before. Then, its uncertainty was calculated considering the uncertainties at the measured flow rate (ISO 5167, 1980), inlet and outlet temperatures and also at the estimated water heat capacity, as given in the following equation (Figueroa and Bensley, 1991):

\[ S = \sqrt{\left(\frac{\partial P}{\partial Q}\right)^2 + \left(\frac{\partial P}{\partial \Delta T}\right)^2 + \left(\frac{\partial P}{\partial \rho}\right)^2} \]  \hspace{1cm} (13)

The uncertainties shown in Table 1 were calculated by the data acquisition software considering all these parameters.

![Steady State Temperatures](image)

**Figure 3. Evolution of the temperatures during the period of stability.**

<table>
<thead>
<tr>
<th>Time [s]</th>
<th>Temperature [°C]</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
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<tr>
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<td>35</td>
</tr>
</tbody>
</table>

![Thermal Power](image)

**Figure 4. Evolution of the thermal power obtained during the period of stability.**

<table>
<thead>
<tr>
<th>Table 1: Results of the Thermal Power Calibration.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average primary loop coolant flow rate:</td>
</tr>
<tr>
<td>Average primary loop inlet temperature:</td>
</tr>
<tr>
<td>Average primary loop outlet temperature:</td>
</tr>
<tr>
<td>Power dissipated in the primary loop:</td>
</tr>
<tr>
<td>Thermal losses from the reactor pool:</td>
</tr>
<tr>
<td>Conduction through the wall:</td>
</tr>
<tr>
<td>Conduction through the bottom:</td>
</tr>
<tr>
<td>Evaporation:</td>
</tr>
<tr>
<td>Convection:</td>
</tr>
<tr>
<td>Total:</td>
</tr>
<tr>
<td>Total reactor power:</td>
</tr>
<tr>
<td>Standard deviation of the readings:</td>
</tr>
<tr>
<td>Uncertainty in the measure of the reactor thermal power:</td>
</tr>
<tr>
<td>Power dissipated in the secondary loop:</td>
</tr>
</tbody>
</table>

**Conclusion**

The power of the TRIGA Mark I IPR-RI Nuclear Reactor at CDTN/CNEN was recently increased from 100 kW to 250 kW. The reactor thermal power calibration is very important for precise neutron flux and fuel element burnup calculations. The burnup is linearly dependent on the reactor thermal power and its accuracy is important to the determination of the mass of burned \(^{238}\)U, fissile products, fuel element activity, decay heat power generation and radiotoxicity.

The calibration method used consisted of the steady-state energy balance of the primary cooling loop. For this balance, the inlet and outlet temperatures and the water flow in this primary cooling loop were measured. The heat transferred through the primary loop was added to the heat leakage from the reactor pool. The thermal losses from the primary loop were not evaluated since the inlet and outlet temperatures were measured just above the water surface of the reactor pool. To minimize the heat leakage the temperature of the water in the reactor pool, as well as the reactor room temperature, were set as close as possible to the soil temperature, since leakage is mainly due to the conduction through the concrete and metal walls to the soil and also due to evaporation and convection through the water surface of the reactor pool.

The heat balance method is accurate, but impractical for monitoring the instantaneous reactor power level, particularly during transients. The power is monitored by four nuclear detectors, which are calibrated by the thermal method described here. This is now the standard procedure for calibrating the power of the TRIGA Mark I IPR-RI Nuclear Reactor. The values calculated for the uncertainties agree with international results (Caglar et al., 1999).

**Acknowledgements**

The authors thank the operation staff of the TRIGA Mark I IPR-RI Nuclear Reactor, for the readiness and dedication in the operation of the reactor during these measurements.

**References**


Souza, R. M. G. P. et al., 2002 “Resultados dos Testes Fisicos para o Aumento de Potência do Reactor TRIGA IPR-R1”, (CNEN/COTN NI - IT4-0702), Belo Horizonte, 54 p.


ON-LINE MEASUREMENT OF THE REACTIVITY TEMPERATURE COEFFICIENT OF THE IPR-R1 TRIGA NUCLEAR RESEARCH REACTOR

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Abstract. On-line monitoring of several new process variables of the IPR-R1 TRIGA Reactor of the Nuclear Technology Development Center – CDTN became possible after the data acquisition and processing system implementation and the installation of one instrumented fuel rod in the reactor core. Several neutronic and thermo-hydraulic parameters are now registered, such as the operation power, the reactivity insertion in the core, the control rod position, the fuel and the water temperatures, and so on. Since the inherently safe operation of a reactor is dependent on the reactivity control, it is essential to have information about this parameter over many different temperature ranges. The fuel elements have been designed to provide a significant prompt negative temperature coefficient that allow safe reactor operation. The developed monitoring system gives, in real time, the reactivity temperature coefficient. The system gives also other reactor parameters such as: reactivity worth of the control rods, when the rod is inserted or withdrawn in the core and also the effect of reactivity during the reactor operation. This paper describes the methodology and the results in on-line monitoring of the temperature coefficient of the IPR-R1 TRIGA Reactor.

Keywords: TRIGA nuclear reactor, instrumented fuel element, reactivity, temperature, core.

1. INTRODUCTION

Reactivity is the most important parameter in nuclear reactor operation. When it is positive the reactor is supercritical, zero at criticality, and negative the reactor is subcritical. Reactivity can be controlled in various ways: by adding or removing fuel; by changing the fraction of neutrons that leaks from the system; or by changing the amount of an absorber that competes with the fuel for neutrons. In a nuclear reactor, temperature changes can introduce reactivity changes. This property is called the "temperature coefficient of reactivity." In water-cooled nuclear reactors, the predominant reactivity changes are brought about by changes in the temperature of the coolant water. In this case the temperature coefficient is negative, which means that an increase in coolant temperature causes a decrease in reactivity, and vice-versa. A reactor with a negative temperature coefficient of reactivity is therefore inherently self-controlling and safe. The TRIGA reactor (Training, Research, Isotopes, General Atomics) uses uranium-zirconium hydride (UZnH) fuel, which has a large, prompt negative thermal coefficient of reactivity, meaning that as the temperature of the core increases, the reactivity rapidly decreases — so it is highly unlikely, though not impossible for a meltdown to occur.

The IPR-R1 TRIGA Nuclear Research Reactor, shown in Fig. 1, is a pool type reactor cooled by natural circulation. The core consists of a lattice of cylindrical fuel-moderator elements and graphite elements. The 250 kW core configuration has 63 fuel elements composed of 58 original aluminum clad elements and 5 fresh stainless steel clad fuel elements. The elements are arranged in five concentric rings, and the spaces between the rods are filled with water that acts as coolant and moderator. The power level of the reactor is controlled with three control rods: a Regulating rod, a Shim rod, and a Safety rod. Fuel temperature was obtained through the use of an instrumented fuel element with thermocouples embedded in the zirconium centerline pin. Fuel temperature measurements were taken in the position B1 (ring B). The inlet and outlet coolant temperatures were measured by using two type K thermocouples inserted in two channels in the core, close to the position B1. A schematic view of the present core configuration is shown in Figure 2. TRIGA reactor utilizes solid fuel elements in which the zirconium hydride moderator is homogeneously combined with 20 % enriched uranium (\(^{235}\text{U}\)). The feature of these fuel-moderator elements is the prompt negative temperature coefficient of reactivity, which automatically limits the reactor power to a safe level in the event of a power excursion. Because of this coefficient, a significant amount of reactivity is needed to overcome the temperature and allow the reactor to operate at high power levels. During steady-state operation, the reactivity in the reactor core is controlled by three independent control rods and drives.
Nuclear reactors must have sufficient excess reactivity to compensate the negative reactivity feedback effects such as those caused by the fuel temperature and power defects of reactivity, fuel burnup, fission poisoning production, and also to allow full power operation for predetermined period of time. To compensate for this excess reactivity, it is necessary to introduce an amount of negative reactivity into the core which one can adjust or control at will. In the IPR-R1 reactor, the reactivity control is done by three control rods that can be inserted into or withdrawn from the core.

The data acquisition system used in the IPR-R1 Reactor consolidates the information about the reactor status and provides an on-line data analysis (Mesquita e Rezende, 2004). The data acquisition program attends to the recommendations of the International Atomic Energy Agency (IAEA, 2002). It will be shown here the methodology used to find the equations that were used in the data acquisition program to monitor, in real-time, the control rods worth, the reactor temperature coefficient of reactivity and the loss of reactivity during the reactor operation.

2. CONTROL RODS WORTH

The determination of the reactivity worth of individual control elements and the effects of such elements on the power distribution in the core is important to the safe and efficient operation of a nuclear reactor. Once a control rod is calibrated, it is possible to evaluate the magnitude of other reactivity changes by comparing the critical rod positions before and after the change. All three control rods are calibrated by the positive period method. The method consists of withdrawing the control rod from a known critical position through a small distance. This adds a positive reactivity to the system and the reactor power increases in an exponential manner with time, and establishes a stable period that is measured using the doubling time, that is the time required for the power to increase by a factor of two. Each successive step is compensated by lowering the other control rod just enough to reestablish criticality. The reactivity associated with the measurement is gotten from the graphical form of the inhour equation, that gives the relationship between reactivity and the stable reactor period.

The experimental data obtained in (Souza et al., 2004), and the integral fitted worth curves of the Regulating, Shim and Safety control rods as a function of their positions are shown graphically in Fig. 3, Fig. 4 and Fig. 5, respectively. The equations representing the fitted model, and the coefficients of determination $R^2$, that confirm the goodness of the fit are also shown in the figures. The integral control rod worth curve is particularly important in research reactor operation. The equations were added to the data acquisition program.
Figure 2. Core configuration of the IPR-R1 TRIGA Reactor

![Core Configuration Diagram]

Figure 3. Reactivity as function of insertion of Regulation control rod

![Reactivity Graph]

\[ y = -2.07 \times 10^{-6} x^3 + 3.21 \times 10^{-3} x^2 - 8.39 \times 10^{-1} x + 7.81 \times 10^1 \]

\[ R^2 = 0.998 \times 10^{-1} \]
3. THE OVERALL TEMPERATURE COEFFICIENT OF REACTIVITY

The prompt temperature coefficient of reactivity is a very important safety parameter of research reactors, it is defined as the change in reactivity for a unit change in the fuel system temperature. A negative temperature coefficient of reactivity is desirable since it tends to counteract the effects of transient temperature changes during reactor operation. In TRIGA reactors the moderator is the hydrogen that is mixed with the fuel itself. If the fuel temperature increases when the control rods are suddenly removed, the neutrons inside the hydrogen-containing fuel rod become warmer than the neutrons outside in the cold water. These warmer neutrons inside the fuel cause less fissioning in the fuel and escape into the surrounding water. The end result is that the reactor automatically reduces the power within a few thousandths of a second, faster than any engineered device can operate. The inherent safety of the TRIGA reactor arises from the prompt negative temperature reactivity coefficient, whose measured value was (-1.1 ± 0.2) $\psi$/°C (Souza et al., 2002), which effectively limits the power when excess reactivity is suddenly inserted.

The overall temperature reactivity coefficient of the reactor refers to the change in the total core temperature. Fuel temperatures were measured by three thermocouples in the center of the instrumented fuel element at location B1. This location is the hottest position in the core. To obtain the overall temperature coefficient it is necessary to know the average temperature in the core. This value was found using the temperature distribution in the core shown in Fig. 6 (Mesquita, 2005). The average temperature as function of the maximum temperature in the core rings follows the equation shown in Fig. 7. The axial temperature distribution in the fuel follows the same distribution of neutron flux, maximum/average = 1.25 (Fig. 8). The average radial temperature distribution inside the fuel, in several operation power, is approximately 1.11 (Fig. 8). The equations found were added to the data acquisition program.
Figure 6. The IPR-R1 temperature distribution in the core rings

Figure 7. Core average temperature as function of core maximum temperature

Figure 8. Experimental axial and radial fuel rod temperature profile
In order to obtain the power effects on the reactivity in the reactor it was performed the following experiment. The reactor power was increased, and, consequently, the fuel temperature, by withdrawing the Shim control rod in steps. All other control rods were completely withdrawn. The power increased with each increasing step, then reached a new, steady, higher level. The reactivity was determined from the calibrated Shim rod curve (Fig. 4), considering each critical rod position. The forced reactor cooling system was not operating during the experiment, and the initial fuel and water temperature at zero power was 24 °C. Table 1 presents the experimental results, and Fig. 9 shows the curve and equation of the total temperature reactivity coefficient versus the core average temperature.

<table>
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<tr>
<th>Reactor Power (kW)</th>
<th>Fuel Temp. Max. (°C)</th>
<th>Core Temp. Average (°C)</th>
<th>ΔT (°C)</th>
<th>Δρ (cents)</th>
<th>Δρ/ΔT (cents/°C)</th>
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<td>166.4</td>
<td>1.5</td>
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</tr>
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</table>

**Figure 9. Overall temperature reactivity coefficient**

The equations of the control rods reactivity as function of their positions in the core, and the core reactivity as function of the temperature and the operation time were added to the data acquisition program. Figure 12 shows the acquisition system screen where the operator can monitor, during the reactor operation, the consolidated reactivity information.
4. CONCLUSIONS

The control of reactivity is one of the most important items that must be performed to ensure the safe and efficient operation of a nuclear research reactor. The reactor operators need to know, in real-time, the basic reactor behavior in order to understand and safely operate a nuclear reactor.

The data acquisition system has been designed and developed to automatically monitor and record all operational parameters of the IPR-R1 TRIGA Reactor. The color monitor provides on-line information about important operating parameters such as: the control rods positions; the control rods worth; the reactivity inserted in the core; the loss of reactivity caused by the fuel temperature, the reactor operation graphics, etc. Hard copies of the displays can be made using the graphics printer. The records of the reactor process variables are important for immediate or subsequent safe analyze, and for reporting the reactor operations to the organization and to external authorities (IAEA, 1995). The system does not propose to control the reactor operation, but to help the operator to get more information about the safety status of systems, and, if necessary, to be used to identify manual actions. The data acquisition and processing system implemented in the IPR-R1 TRIGA Reactor is the beginning of the control and instrumentation update to this reactor. In the future all the reactor operation will be made automatically by programmable logical controllers (PLC’s), like other modern research and power reactors (Mizuki et al, 1995 and Swaminathan, 2005).

The overall temperature coefficient of reactivity presented in this work is a preliminary result. The uncertainty of this parameter is about ± 15%, mainly due to the uncertainty in power calibration of the reactor, which estimated value is ± 7.2% (Mesquita et al., 2007).
3. ACKNOWLEDGEMENTS

The author would like to express their special thanks to the IPR-R1 TRIGA operator team for their help during the experiments and the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) and Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq) for the partial financial support.

4. REFERENCES


Souza, R.M.G.P. et al, 2002, “Resultados dos Testes Finais para o Aumento de Potência do Reator TRIGA IPR-R1”, NI-IT4-07/02, CDTN/CNEN, Belo Horizonte, Brazil.


5. RESPONSIBILITY NOTICE

The authors are the only responsible for the printed material included in this paper.
On-line monitoring of the IPR-R1 TRIGA reactor

neutronic parameters

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ABSTRACT

The on-line monitoring of several new process variables of the IPR-R1 TRIGA Reactor of the Nuclear Technology Development Center – CDTN became possible after the data acquisition and processing system implementation and the installation of one instrumented fuel rod in the reactor core. Several neutronics and thermo-hydraulics parameters are now registered, such as the operation power, the reactivity insertion in the core, the control rod position, the fuel and the water temperatures, and so on. Since the inherently safe operation of a reactor is dependent on the reactivity control, it is essential to have information on this parameter over many different temperature ranges. The fuel elements have been designed to provide a significant negative prompt temperature coefficient that allows safe reactor operation. The developed monitoring system gives the reactivity worth of the control rods, when the rod considered is inserted into the core or withdrawn from it, and also the loss of reactivity during the reactor operation. This paper describes the methodology and the results found with the on-line monitoring of the reactivity behavior of the IPR-R1 TRIGA Reactor.

Keywords: TRIGA research reactor, reactivity, control rods worth, fuel temperature.

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1. INTRODUCTION

The IPR-R1 TRIGA Nuclear Research Reactor is a pool type reactor cooled by natural circulation, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in $^{235}$U.

Nuclear reactors must have sufficient excess reactivity to compensate the negative reactivity feedback effects such as those caused by the fuel temperature and power defects of reactivity, fuel burnup, fission poisoning production, and also to allow full power operation for predetermined period of time. To compensate for this excess reactivity, it is necessary to introduce an amount of negative reactivity into the core which one can adjust or control it at will. In the IPR-R1 Reactor the reactivity control is done by three control rods that can be inserted into or withdrawn from the core.

The data acquisition system used in the IPR-R1 Reactor consolidates information about the reactor status and provides an on-line data analysis (Mesquita and Rezende, 2004). The data acquisition program responds to the recommendations of the International Atomic Energy Agency - IAEA (2002). It will be shown here the methodology used to find the equations that were used in the data acquisition program to monitor, in real-time, the control rods worth, the reactor temperature coefficient of reactivity and the loss of reactivity during the reactor operation.

2. CONTROL RODS WORTH

The knowledge of the reactor's response to specific control rod motions is essential to the safe and efficient operation of a nuclear reactor. The effectiveness, or worth, of a control rod depends largely upon the value of the neutron flux at the location of the rod. Figure 1 presents a top view of the IPR-R1 core configuration, and the Regulating, Shim and Safety control rod positions.
All three-control rods were calibrated by the positive period method that consists of withdrawing the control rod from a known critical position through a small distance, and then to measure the stable period of the resultant reactor transient. The period was obtained using the doubling time, that is the time required for the power to increase by a factor of two. Each successive step is compensated by lowering the other control rod just enough to reestablish criticality. In this process the control rod under calibration proceeds from the most inserted position (maintaining the reactor critical) to fully removed. The Safety and Shim rods were intercalibrated. The idea was to measure one control rod in presence of another rod, used for compensating the reactivity introduced by step withdrawal of the measure rod. The reactivity measurements were performed at a low power so the temperature increase during the experiment was negligible. The reactivity values associated with the periods obtained were gotten from the graphical form of the inhour equation.

The experimental data obtained in (Souza et al, 2002), and the integral fitted worth curves of the Regulating, Shim and Safety control rods as a function of their positions are shown graphically in Fig. 2, Fig. 3 and Fig. 4, respectively. The equations representing the fitted model, and the coefficients of determination $R^2$, that confirm the goodness of the fit, are also shown in the figures. The equations were added to the data acquisition program. The integral control rod worth curve is particularly important in research reactor operation. The measured values of the Regulating, Shim and Safety control rod worth were 0.5, 3.1 and 2.8 cents, respectively (Souza et al, 2002).

3. COEFFICIENTS OF REACTIVITY

Temperature is one of the operating conditions that affect the reactivity of a reactor core. Such reactivity variation with temperature is the principal feedback mechanism.
determining the inherent stability of a nuclear reactor. The temperature coefficient of reactivity is defined as the change in reactivity due to a variation in the average temperatures of each component of the core. A negative temperature coefficient of reactivity is desirable since it tends to counteract the effects of transient temperature changes during reactor operation. An increase in temperature will cause a decrease in the reactivity, hence a decrease in reactor power and temperature which tends to stabilize the reactor power level.

In TRIGA reactors the moderator is the hydrogen that is mixed with the fuel itself. If the fuel temperature increases when the control rods are suddenly removed, the neutrons inside the hydrogen-containing fuel rod become warmer than the neutrons outside in the cold water. These warmer neutrons inside the fuel cause less fissioning in the fuel and escape into the surrounding water. The end result is that the reactor automatically reduces the power within a few thousandths of a second, faster than any engineered device can operate. The inherent safety of the TRIGA reactor arises from the negative prompt temperature reactivity coefficient, whose measured value was (-1.1 ± 0.2) e/°C (Souza and Resende, 2004). The prompt temperature coefficient refers only to the fuel temperature, and the overall temperature coefficient of the reactor refers to the change in the total core temperature.

Fuel temperatures were measured by three thermocouples in the center of the instrumented fuel element at location B1, which is the hottest position in the core. To obtain the overall temperature coefficient it is necessary to know the average temperature in the core. This value was found using the temperatures distribution in the core shown in Fig. 5 (Mesquita, 2005). The average temperature as function of the maximum temperature in the core rings follows the equation shown in Fig. 6. The axial temperature distribution in the fuel follows the same distribution of the neutron flux, maximum/average = 1.25 (Fig. 7), and the radial temperature distribution inside the fuel, in several operation power, is approximately 1.11 (Fig. 7).
In the reactivity experiment, performed in (Souza et al, 2006), the reactor power was increased, and, consequently, the fuel temperature and the core temperature, by withdrawing the shim control rod in steps. All other control rods were completely withdrawn. The power increased with each increasing step, then reached a new, steady, higher level. The reactivity was determined from the calibrated shim rod curve (Fig. 3), considering each critical rod position. The forced reactor cooling system was not operating during the experiment, and the initial fuel and water temperatures at zero power were 24 °C. Figure 8 shows the experimental curve and equation of the total temperature reactivity coefficient versus the core average temperature, and the core reactivity evolution as a function of the fuel temperature is presented in Fig. 9.

Figure 10 shows the power coefficient of reactivity as a function of the reactor power level, and Fig. 11 presents the associated reactivity loss to achieve a given power level (Souza et al, 2006). In the last figure the curve is almost linear and gives a power coefficient of, approximately, -0.65 cent/kW. Because of the prompt negative temperature coefficient, a significant amount of reactivity is needed to overcome temperature and allow the reactor to operate at higher power levels in steady state operation. The power defect, that is the change in reactivity taking place between zero power and full power, is around 1.8 

The purpose of these experiments were to measure the core temperature reactivity coefficient as a function of the core temperature and the loss of reactivity as a function of the fuel temperature and the reactor power. The equations found in were added to the data acquisition program.

4. FISSION PRODUCT POISONING

During the operation of a nuclear reactor many fission products are generated. The xenon-135 is the most significant of them because it acts as poison in the reactor, affecting the reactivity of the core, due its enormous thermal neutron absorption cross section. Figure 12 shows the loss of reactivity caused by the $^{135}\text{Xe}$ poisoning during the reactor operation at 100 kW, and Fig. 13 shows the same loss at 250 kW (Souza et al, 2002). The two regression equations, added to the data acquisition program, and their coefficients of determination ($R^2$) are given in the figures.

Figure 14 shows the acquisition system screen where the operator can monitoring, during the reactor operation, the consolidated reactivity information.

5. CONCLUSIONS

The reactivity control is one of the most important items that must be performed to ensure the safe and efficient operation of a nuclear research reactor. The reactor operators need to know, in real-time, the basic reactor behavior in order to understand and safely operate a nuclear reactor.

The data acquisition system has been designed and developed to automatically monitor and record the operational parameters of the IPR-R1 TRIGA Reactor. The color monitor provides on-line information about important operating parameters
such as: the control rods positions; the control rods worth; the reactivity inserted in the core; the loss of reactivity caused by the xenon poisoning and the fuel temperature; the reactor operation graphics, etc. Hard copies of the displays can be made using the graphics printer. The records of the reactor process variables are important for immediate or subsequent safe analyze, and for reporting the reactor operations to the organization and to external authorities (IAEA, 2002). The system does not propose to control the reactor operation, but to help the operator to get more information about the safety status of the systems, and, if necessary, to be used to identify manual actions.

The data acquisition and processing system implemented in the IPR-R1 TRIGA Reactor is the beginning of the control and instrumentation update of this reactor. In the future all the reactor operation will be made by programmable logical controllers (PLC’s), like other modern research and power reactors (Hai et al, 2003), (Mizuki et al, 1995) and (Swaminathan, 2005).

The overall temperature coefficient of reactivity presented in this work is a preliminary result. The uncertainty of this parameter is about ± 15%, mainly due to the uncertainty in power calibration of the reactor, which estimated value is ± 7.2% (Mesquita et al, 2007).

ACKNOWLEDGMENTS

The authors express their thanks to the IPR-R1 operational staff and to the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the partial financial support.
Fig. 1. Core configuration of the IPR-R1 TRIGA Reactor.
Fig. 2. Experimental integral Regulating control rod worth curve.
Fig. 3. Experimental integral Shim control rod worth curve.
Fig. 4. Experimental integral Safety control rod worth curve.
Fig. 5. The IPR-R1 temperature distribution in the core rings.
Fig. 6. Core average temperature as a function of core maximum temperature.
Fig. 7. Experimental axial and radial fuel rod temperature profiles.
Fig. 8. Overall temperature reactivity coefficient.
Fig. 9. Change in reactivity as a function of fuel temperature.

$y = -4.33E-06x^4 + 1.25E-05x^3 - 4.14E-04x^2 - 5.96E-01x + 1.74E+01$

$R^2 = 1.00$
Fig. 10. Power coefficient of reactivity versus reactor power level.
Fig. 11. Change in reactivity as a function of reactor power.
Fig. 12. Xenon poisoning during power operation at 100 kW.
Fig. 13. Xenon poisoning during power operation at 250 kW.
Fig. 14. Reactivity monitoring on the screen of the data acquisition system.
REFERENCES


Thermal methods for on-line power monitoring of the

IPR-R1 TRIGA Reactor

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ABSTRACT

The IPR-R1 Research Reactor is a TRIGA Mark I type reactor. It is a pool reactor, and the fuel elements are cooled by water natural convection. The heat removal capacity of this process is great enough for safety reasons at the current maximum 250 kW reactor power level. However, a heat removal system is provided for removing heat from the reactor pool water. Power monitoring of nuclear reactors is always done by means of neutronic instruments. In the IPR-R1 the power is measured by four nuclear channels. This work presents the results and methodology for monitoring the power of this reactor by three thermal processes. One method uses the temperature difference between an instrumented fuel element and the pool water below the reactor core. The other two methods consist in the steady-state energy balance of the primary and secondary reactor cooling loops.

Keywords: Research reactors, TRIGA reactor, instrumented fuel rod, power, temperature.

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6. INTRODUCTION

The IPR-R1 reactor fuel is an alloy of zirconium hydride and uranium enriched at 20% in $^{235}$U. Figure 1 shows the reactor pool and Fig. 2 shows the core top view. The reactor core has 58 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements located in five rings around the central thimble. One of these stainless steel-clad fuel elements is instrumented with three thermocouples along its centerline in order to evaluate the reactor thermal hydraulic performance (Mesquita, 2005).

Fuel temperatures were measured in different power levels by an instrumented fuel element at location B1, which is the hottest position in the core. Three new processes for reactor power measurement by thermal ways were developed as a result of the experiments. These processes make it possible on-line or off-line evaluation of the reactor power and the analysis of its behavior.

7. POWER MEASURING CHANNELS USING NEUTRONIC METHODS

Power monitoring of nuclear reactors is always done by means of nuclear detectors, which are calibrated by thermal methods. In the IPR-R1 Reactor four neutron-sensitive chambers are mounted around the reactor core for flux measurement. The departure channel consists of a fission counter with a pulse amplifier that a logarithmic input rate circuit. The logarithmic channel consists of a compensated ion chamber, whose signal is the input to a logarithmic amplifier, which gives a logarithmic power indication from less than 0.1 W to full power. The linear channel consists of a compensated ion chamber, whose signal is the input to a sensitive amplifier and recorder with a range switch, which gives accurate power information from source level to full power on a linear recorder. The percent channel consists of an uncompensated ion chamber, whose
signal is the input to a power level monitor circuit and meter, which is calibrated in percentage of full power. Finally, the ionization chamber neutron detector measures the flux of neutrons thermalized in the vicinity of the detector. This signal is not always proportional to the integral neutron flux in the core and consequently to the core power. Besides, the response of a single nuclear detector is sensitive to the changes in the core configuration, particularly to the control rod position. This is important in the TRIGA reactor, which do not have distributed absorbers for reactivity control and criticality maintenance is doing by insertion of control rods (Zagar et al, 1999).

8. POWER MEASURING CHANNELS USING THERMAL PROCESS

3.1 Power Measuring Channel by Thermal Balance

The reactor core is cooled by natural convection of demineralized light water in the reactor pool. Heat is removed from the reactor pool and released into the atmosphere through the primary cooling loop, the secondary cooling loop and the cooling tower (Fig. 3). Pool temperature depends on reactor power, as well as external temperature, because the latter affects heat dissipation in the cooling tower. The total power is determined by the thermal balance of cooling water flowing through the primary and secondary loops added to the calculated heat losses. These losses represent a very small fraction of the total power (about 1.5% of total) (Mesquita et al, 2007).

The inlet and outlet temperatures are measured by four platinum resistance thermometers (PT-100) positioned at the inlet and at the outlet pipes of the primary and secondary cooling loops. The flow-rate in the primary loop is measured by an orifice plate and a differential pressure transmitter. The flow in the secondary loop is measured by a flowmeter. The pressure transmitter and the temperature measuring lines were
calibrated and an adjusted equation was added to the data acquisition system (Mesquita and Rezende, 2004).

The steady state is reached after some hours of reactor operation, so that the power dissipated in the cooling system added with the losses should be equal to the core power. The uncertainty in the power measurement considered all the uncertainty propagation from primary parameters, according to the methodology described by Coleman and Steele (1999).

The thermal power dissipated in the primary and secondary loops were given by:

\[ q_{\text{cool}} = \dot{m} \cdot c_p \cdot \Delta T, \]  \hspace{1cm} (1)

where \( q_{\text{cool}} \) is the thermal power dissipated in each loop [kW], \( \dot{m} \) is the flow rate of the coolant water in the loop [kg/s], \( c_p \) is the specific heat of the coolant [kJ/kg°C], and \( \Delta T \) is the difference between the temperatures at loop the inlet and outlet [°C]. The data acquisition computer program calculates the power dissipated in the cooling loop with the collected data being used in Equation 1, and with the \( \dot{m} \) and \( c_p \) values calculated as function of coolant temperature (Miller, 1989).

To calculate the heat losses, one resistance thermometer (PT-100) was positioned inside the pool to measure the water pool temperature. A type K thermocouple was placed just above the pool surface to measure the air temperature at the reactor room. Two type K thermocouples were distributed around the pool, in holes in the reactor room floor, to measure the soil temperature. The core of the TRIGA Mark I IPR-R1 Nuclear Reactor is placed below the room floor, in the bottom of a cylindrical pool, 6.625 m deep and 1.92 m in diameter, whose upper surface is 25 cm below the level of the floor. The reactor pool transfers heat to the environment by conduction to the soil,

272
through the lateral walls and through the bottom of the pool, and by convection and evaporation to the air at the reactor room, through the upper surface. All these losses are calculated by the data acquisition system as described by Mesquita et al. (2007). Figure 4 shows the power evolution in the primary and secondary loops during one reactor operation.

3.2. Power Measuring Channel by Fuel and Pool Temperature

To evaluate the thermal hydraulic performance of the IPR-R1 Reactor one instrumented fuel element was put in the core for the experiments (Mesquita, 2005). The instrumented fuel is identical to standard fuel elements but it is equipped with three chromel-alumel thermocouples, embedded in the zirconium pin centerline. The sensitive tips of the thermocouples are located one at the center of the fuel section and the other two 25.4 mm above, and 25.4 mm below the center. Figure 5 shows the diagram and design of the instrumented fuel element (Gulf General Atomic, 1972). We can see the instrumented fuel element in the center of the core upper view shown in the Figure 2.

During the experiments it was observed that the temperature difference between fuel element and the pool water below the reactor core (primary loop inlet temperature) do not change for the same power value as can been seen on Fig. 6. With the instrumented fuel element in the hottest fuel element of the core (position B1), the power measured in linear channel (with the values corrected by the calibration results) was plotted as a function of the temperature difference between the fuel and the primary loop inlet temperature. The following polynomial expression was obtained that relates the two values:

\[ q = 2 \cdot 10^{-5} (\Delta T)^3 - 0.0045(\Delta T)^2 + 0.7666 \Delta T - 2.4475 \]  

(2)
where \( q \) is the calibrated reactor thermal power, in [kW] and \( \Delta T \) is the difference between the average fuel temperature and the primary loop inlet temperature, in [°C]. It was obtained a determination coefficient equal to one \( (R^2 = 1) \). Equation 2 was included in the data acquisition system and this new power measurement channel is available for the IPR-R1 TRIGA Reactor. After the experiments the instrumented fuel element was maintained in position B1 of the core to monitor the reactor power and core temperature in all reactor operation. Figure 7 compare the reactor power measuring results using the linear neutron channel and the temperature difference channel method. There is a good agreement between the two results, although the temperature difference method presents a delay in its response, and it is useful for steady-state or very slow transients.

The fuel temperature limit defined in the IPR-R1 TRIGA Reactor Accidents Report (CDTN/CNEN, 2007) during steady state operation is 550 °C. A power operation limit over 1 MW was found based in this temperature and using Equation 2.

Figure 8 shows one of the video-screens displays of the digital monitoring system computer that consolidates information of the reactor power status in real time. This screen monitors the power measured by the neutronics channels and by the three new thermal channels (Mesquita and Rezende, 2004).

9. CONCLUSIONS

The knowledge of the reactor thermal power is very important for precise neutron flux and fuel element burnup calculations. The burnup is linearly dependent on the reactor thermal power and its accuracy is important to the determination of the mass of burned \(^{235}\)U, fission products, fuel element activity, decay heat power generation and radiotoxicity. The thermal balance method presented in this report is now the standard
methodology used for the IPR-R1 TRIGA Reactor power calibration (CDTN/CNEN, 2007). The uncertainty value obtained does not differ significantly from another thermal calibration processes described in technical literature (Zagar et al, 1999). The heat balance and fuel temperature methods are accurate, but impractical methods for monitoring the instantaneous reactor power level, particularly during transients. For transients the power is monitored by the nuclear detectors, which are calibrated by the thermal balance method. On the other hand, the response of one nuclear detector is sensitive to the changes in the core configuration, mainly to the control rod position. This is important in research reactors, which do not have distributed absorbers for reactivity control and the normal mode of maintaining criticality is by insertion of control rods. The heating of the thermocouple due to the gamma ray is negligible because the small mass and good thermal radiation equilibrium with the surrounding fuel.

ACKNOWLEDGMENTS

The authors express their thanks to the IPR-R1 operational staff and to the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the partial financial support.

REFERENCES


Fig. 1. The IPR-R1 TRIGA Research Nuclear Reactor.
Fig. 2. Top view of the IPR-R1 TRIGA core.
Fig. 3. The reactor cooling system.
Fig. 4. Thermal power evolution in the cooling system.
Fig. 5. Diagram of the instrumented fuel element.
Fig. 6. Fuel and inlet core water temperatures evolution.
Fig. 7. Reactor power measured by neutron channel and by fuel element temperature.
Fig. 8 Power monitoring on the screen of the data acquisition system.
ON-LINE MONITORING OF THE REACTIVITY AND CONTROL RODS WORTH AT THE IPR-R1 TRIGA REACTOR

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ABSTRACT

On-line monitoring of several new process variables of the IPR-R1 TRIGA Reactor of the Nuclear Technology Development Center – CDTN became possible after the data acquisition and processing system implementation and the installation of one instrumented fuel rod in the reactor core. Several neutronics and thermo-hydraulics parameters are now registered, such as the operation power, the reactivity insertion in the core, the control rod position, the fuel and the water temperature, and so on. Since the inherently safe operation of a reactor is dependent on the reactivity control, it is essential to have information on this parameter over many different temperature ranges. The fuel elements have been designed to provide a significant prompt negative temperature coefficient that allow safe reactor operation. The developed monitoring system gives the reactivity worth of the control rods, when the rod considered is inserted or withdrawn in the core and also the loss of reactivity during the reactor operation. This paper describes the methodology and the results found with in the on-line monitoring of the reactivity behavior of the IPR-R1 TRIGA Reactor.

1. INTRODUCTION

The IPR-R1 TRIGA Nuclear Research Reactor is a pool type reactor cooled by natural circulation, and having as fuel an alloy of zirconium hydride and uranium enriched at 20% in $^{235}\text{U}$. Nuclear reactors must have sufficient excess reactivity to compensate the negative reactivity feedback effects such as those caused by the fuel temperature and power defects of reactivity, fuel burnup, fission poisoning production, and also to allow full power operation for predetermined period of time. To compensate for this excess reactivity, it is necessary to introduce an amount of negative reactivity into the core which one can adjust or control it at will. In the IPR-R1 Reactor the reactivity control is done by three control rods that can be inserted into or withdrawn from the core.

The data acquisition system used in the IPR-R1 Reactor consolidates information about the reactor status and provides an on-line data analysis [1]. The data acquisition program responds to recommendations of the International Atomic Energy Agency - IAEA [2]. It will be shown here the methodology used to find the equations that were used in the data acquisition program to monitor, in real-time, the control rods worth, the reactor temperature coefficient of reactivity and the loss of reactivity during the reactor operation.
2. CONTROL RODS WORTH

The effectiveness, or worth, of a control rod depends largely upon the value of the neutron flux at the location of the rod. The determination of the control rods reactivity is a very important aspect of nuclear reactor core design. All three-control rods were calibrated by the positive period method. The method consists of withdrawing the control rod from a known critical position through a small distance. Each successive step is compensated by lowering the other control rod just enough to reestablish criticality. In this process the control rod under calibration proceeds from the most inserted position (maintaining the reactor critical) to fully removed. The Safety and Shim rods were intercalibrated. The idea was to measure one control rod in presence of another rod, used for compensating the reactivity introduced by step withdrawal of the measure rod. The reactivity measurements were performed at a low power so the temperature increase during the experiment was negligible. The period was obtained using the Doubling Time – DT, that is the time required for the power to increase by a factor of two. To obtain the DT it is necessary to wait approximately for 2 minutes, after withdrawing the control rod under calibration, to finish the transition region. The doubling time was measured with two digital chronometers, observing the power showed in the console. The reactivity associated with the measurement was gotten from the graphical form of the Inhour equation [3].

Figure 1 show the IPR-R1 core configuration. The experimental data obtained in [3], and the integral fitted worth curves of the Regulating, Shim and Safety control rods as a function of their positions are shown graphically in Fig. 2, Fig. 3 and Fig. 4. The equations representing the fitted model, and the coefficients of determination $R^2$, that confirm the goodness of the fit are also shown in the figures. The integral control rod worth curve is particularly important in research reactor operation. The equations were added to the data acquisition program.

![Figure 1. Core configuration of the IPR-R1 TRIGA Reactor.](image-url)
Figure 2. Reactivity as function of insertion of Regulation control rod.

Figure 3. Reactivity as function of insertion of Shim control rod.

Figure 4. Reactivity as function of insertion of the Safety control rod.

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3. THE OVERALL TEMPERATURE COEFFICIENT OF REACTIVITY

The temperature coefficient of reactivity is a very important safety parameter of research reactors, it is defined as the change in reactivity for a unit change in the fuel system temperature. A negative temperature coefficient of reactivity is desirable since it tends to counteract the effects of transient temperature changes during reactor operation. In TRIGA reactors the moderator is the hydrogen that is mixed with the fuel itself. If the fuel temperature increases when the control rods are suddenly removed, the neutrons inside the hydrogen-containing fuel rod become warmer than the neutrons outside in the cold water. These warmer neutrons inside the fuel cause less fissioning in the fuel and escape into the surrounding water. The end result is that the reactor automatically reduces the power within a few thousandths of a second, faster than any engineered device can operate. The inherent safety of the TRIGA reactor arises from the prompt negative temperature reactivity coefficient, whose measured value was (-1.1 ± 0.2) °C [4], which effectively limits the power when excess reactivity is suddenly inserted. The prompt temperature coefficient refers only to fuel temperature, and the overall temperature coefficient of the reactor refers to the change in the total core temperature.

The purpose of this experiment was to measure the core temperature reactivity coefficient as function of the core temperature and the loss of reactivity as function of the fuel temperature and reactor power. The equations found were added to the data acquisition program. Fuel temperatures were measured by three thermocouples in the center of the instrumented fuel element at location B1. This location is the hottest position in the core. To obtain the overall temperature coefficient it is necessary to know the average temperature in the core. This value was found using the temperatures distribution in the core shown in Fig. 5 [5]. The average temperature as function of the maximum temperature in the core rings follows the equation shown in Fig. 6. The axial temperature distribution in the fuel follows the same distribution of neutron flux, maximum/average = 1.25 (Fig. 7). The radial temperature distribution inside the fuel, in several operation power, is approximately 1.11 (Fig. 7).

![Figure 5. The IPR-R1 temperature distribution in the core rings [5].](image-url)
Figure 6. Core average temperature as function of core maximum temperature [5].

Figure 7. Experimental axial and radial fuel rod temperature profile [5].

In the reactivity experiment, performed in [6], the reactor power was increased, and, consequently, the fuel temperature by withdrawing the Shim control rod in steps. All other control rods were completely withdrawn. The power increased with each increasing step, then reached a new, steady, higher level. The reactivity was determined from the calibrated Shim rod curve (Fig. 3), considering each critical rod position. The forced reactor cooling system
was not operating during the experiment, and the initial fuel and water temperature at zero power was 24 °C. Table 1 presents the experimental results, and Fig. 8 shows the curve and equation of the total temperature reactivity coefficient versus the core average temperature.

### Table 1. Experimental results

<table>
<thead>
<tr>
<th>Reactor Power (kW)</th>
<th>Fuel Temp. Max. (°C)</th>
<th>Core Temp. Average (°C)</th>
<th>ΔT (°C)</th>
<th>Δρ (cents)</th>
<th>Δρ/ΔT (cents/°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.01</td>
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<td>24.3</td>
<td>0.0</td>
<td>-7.0</td>
<td>-1.40</td>
</tr>
<tr>
<td>5.3</td>
<td>40.8</td>
<td>29.3</td>
<td>5.0</td>
<td>-10.5</td>
<td>-1.29</td>
</tr>
<tr>
<td>17.0</td>
<td>59.2</td>
<td>37.4</td>
<td>8.1</td>
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<td>-0.97</td>
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<tr>
<td>21.2</td>
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<td>41.0</td>
<td>3.6</td>
<td>-14.5</td>
<td>-1.09</td>
</tr>
<tr>
<td>40.3</td>
<td>94.3</td>
<td>54.3</td>
<td>13.3</td>
<td>-16.5</td>
<td>-0.90</td>
</tr>
<tr>
<td>66.8</td>
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<td>72.7</td>
<td>18.4</td>
<td>-27.0</td>
<td>-0.80</td>
</tr>
<tr>
<td>111.3</td>
<td>185.5</td>
<td>106.3</td>
<td>33.7</td>
<td>-26.5</td>
<td>-0.99</td>
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<tr>
<td>154.8</td>
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<td>133.2</td>
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</tr>
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<td>269.4</td>
<td>164.8</td>
<td>4.4</td>
<td>-3.5</td>
<td>-2.31</td>
</tr>
</tbody>
</table>

**Figure 8. Overall temperature reactivity coefficient.**

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Figure 9 shows the core reactivity evolution as a function of the fuel temperature and Fig. 10 presents the associated reactivity loss to achieve a given power level [6].

![Reactivity vs. Fuel Temperature](image1)

Figure 9. Change in reactivity as function of fuel temperature.

![Reactivity vs. Power](image2)

Figure 10. Change in reactivity as function of reactor power.

4. FISSION PRODUCT POISONING

During the course of operation of a nuclear reactor, the fission fragments and their many decay products accumulate. The xenon-135 is the mainly substance, because it has large cross
section for thermal-neutron absorption. Figure 11 shows the loss of reactivity caused by the $^{135}$Xe poisoning during the reactor operation at 100 kW and Fig. 12 shows the same loss at 250 kW [3]. The two regression equations and their coefficient of determination ($R^2$) are given in the figures.

![Figure 11. Xenon poisoning during power operation at 100 kW.](image)

![Figure 12. Xenon poisoning during power operation at 250 kW.](image)

The equations of the control rods reactivity as function of their positions in the core, and the core reactivity as function of the temperature and the operation time were added to the data acquisition program. Figure 13 shows the acquisition system screen where the operator can monitoring, during the reactor operation, the consolidated reactivity information.

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Figure 13. Reactivity monitoring on the screen of the data acquisition system.

5. CONCLUSIONS

The control of reactivity is one of the most important items that must be performed to ensure the safe and efficient operation of a nuclear research reactor. The reactor operators need to know, in real-time, the basic reactor behavior in order to understand and safely operate a nuclear reactor.

The data acquisition system has been designed and developed to automatically monitor and record all operational parameters of the IPR-R1 TRIGA Reactor. The color monitor provides online information about important operating parameters such as: the control rods positions; the control rods worth; the reactivity inserted in the core; the loss of reactivity caused by the xenon poisoning and the fuel temperature; the reactor operation graphics, etc. Hard copies of the displays can be made using the graphics printer. The records of the reactor process variables are important for immediate or subsequent safe analyze, and for reporting the reactor operations to the organization and to external authorities [2]. The system does not propose to control the reactor operation, but to help the operator to get more information about the safety status of systems, and, if necessary, to be used to identify manual actions.
The data acquisition and processing system implemented in the IPR-R1 TRIGA Reactor is the beginning of the control and instrumentation update to this reactor. In the future all the reactor operation will be made by programmable logical controllers (PLC's), like other modern research and power reactors [7], [8] e [9].

The overall temperature coefficient of reactivity presented in this work is a preliminary result. The uncertainty of this parameter is about ± 15%, mainly due to the uncertainty in power calibration of the reactor, which estimated value is ± 7.2% [10].

ACKNOWLEDGMENTS

The authors would like to thank the operation staff of the IPR-R1 TRIGA Research Reactor and the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the financial support.

REFERENCES

ON LINE SYSTEM FOR POWER MONITORING OF THE IPR-R1 TRIGA REACTOR BY THERMAL METHODS

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ABSTRACT

The IPR-R1 Research Reactor is a TRIGA Mark I type reactor. The IPR-R1 is a pool reactor, and the fuel elements at the core are cooled by water natural convection. The heat removal capacity of this process is great enough for safety reasons at the current maximum 250 kW power levels of the reactor. However, a heat removal system is provided for removing heat from the reactor pool water. Power monitoring of nuclear reactors is always done by means of neutronics instruments. In the IPR-R1 the power is measured by four nuclear channels. This work presents the results and methodology for monitoring the power of this reactor by thermal processes. Three improved methods for thermal measuring channels are described using the fuel element temperature and the steady-state energy balance of the primary and secondary cooling loops.

1. INTRODUCTION

The IPR –R1 reactor fuel is an alloy of zirconium hydride and uranium enriched at 20% in $^{235}$U. Figure 1 shows two photographs of the reactor pool and the core. The reactor core has 58 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements located in five rings around the central thimble. One of these steel-clad fuel elements is instrumented with three thermocouples along its centerline. This instrumented fuel element was put in the reactor core in order to evaluate the thermal hydraulic performance of the IPR-R1 reactor [1].

Figure 1. The IPR-R1 TRIGA Research Nuclear Reactor.
Fuel temperatures were measured in various locations throughout the core with the use of the instrumented fuel element at different power levels. Three new processes for reactor power measurement by thermal ways were developed as a result of the experiments. These processes make it possible on-line or off-line evaluation of the reactor power and the analysis of its behavior.

2. POWER MEASURING CHANNELS USING NEUTRONIC METHODS

Power monitoring of nuclear reactors is always done by means of nuclear detectors, which are calibrated by thermal methods. In the IPR-R1 Reactor, four neutron-sensitive chambers are mounted around the reactor core for flux measurement. The departure channel consists of a fission counter with a pulse amplifier that a logarithmic count rate circuit. The logarithmic channel consists of a compensated ion chamber, the signal is an input to a logarithmic amplifier, which gives a logarithmic power indication from less than 0.1 W to full power. The linear channel consists of a compensated ion chamber, the signal is an input to a sensitive amplifier and recorder with a range switch, which gives accurate power information from source level to full power on a linear recorder. The percent channel consists of an uncompensated ion chamber, the signal is an input to a power level monitor circuit and meter, which is calibrated in percentage of full power. Finally, the ionization chamber neutron detector measures the flux of neutrons thermalized in the vicinity of the detector. This signal is not always proportional to the integral neutron flux in the core and consequently to the core power. Besides the response of a single nuclear detector is sensitive to the changes in the core configuration, particularly to the control rod position. This is important in the TRIGA reactor, which does not have distributed absorbers for reactivity control and criticality maintenance is done by insertion of control rods [2].

3. POWER MEASURING CHANNEL USING THERMAL PROCESS

3.1. Power Measuring Channel by Thermal Balance

The reactor core is cooled by natural convection flow of demineralized light water in the reactor pool. Heat is removed from the reactor pool and released into the atmosphere by primary cooling loop, secondary cooling loop and a cooling tower (Fig. 2). Pool temperature depends on reactor power, as well as external temperature, because the latter affects heat dissipation in the cooling tower. The total dissipated power is determined by making the thermal balance of the inlet and outlet cooling water that flows through in the primary and secondary loops and the calculation of the heat losses. These losses represent a very small fraction of the total power (about 1.5% of total) [3].

The inlet and outlet temperatures are measured by four platinum resistance thermometers (PT-100) positioned at the inlet and at the outlet pipes of the primary and secondary cooling loops. The flow rate in the primary loop is measured by an orifice plate and a differential pressure transmitter, in the secondary loop the flow is measured by a flowmeter. The pressure transmitter and the temperature measuring lines were calibrated and an adjusted equation was added to the data acquisition system [4].

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power measurement considered all the uncertainty propagation from primary parameters, according to the methodology described by Coleman and Steele [5].

The thermal power dissipated in the primary and secondary loops were given by:

\[ q_{\text{cool}} = \dot{m} \cdot c_p \cdot \Delta T \]  

(1)

Where \( q_{\text{cool}} \) is the thermal power dissipated in each loop [kW], \( \dot{m} \) is the flow rate of the coolant water in the loop [kg/s], \( c_p \) is the specific heat of the coolant [kJ/kg°C], and \( \Delta T \) is the difference between the temperatures at the inlet and the outlet of the loop [°C]. The data acquisition computer program calculates the power dissipated in the cooling loop with the collected data being used in Equation 1, and with the \( \dot{m} \) and \( c_p \) values corrected as function of coolant temperature [6].

![Diagram of the reactor cooling system](image)

**Figure 2. The reactor cooling system.**

To calculated the heat losses, one resistance thermometer (PT-100) was positioned inside the pool to measure the water pool temperature. A type K thermocouple was placed just above the pool surface to measure the air temperature at the reactor room. Two type K thermocouples were distributed around the pool, in holes in the reactor room floor, to measure the soil temperature. The core of the TRIGA Mark I IPR-R1 Nuclear Reactor is placed below the room floor, in the bottom of a cylindrical pool, 6.625 m deep and 1.92 m in
diameter, whose upper surface is 25 cm below the level of the floor. The reactor pool transfers heat to the environment by conduction to the soil, through the lateral walls and through the bottom of the pool, and by convection and evaporation to the air at the reactor room, through the upper surface. All this losses is calculated by the data acquisition system as described by Mesquita et al. [3]. Figure 3 shows the power evolution in the primary and secondary loops during one reactor operation.

![Thermal Power Dissipated in the Cooling System](image)

**Figure 3.** Thermal power evolution in the cooling system.

3.2. Power Measuring Channel by Fuel and Pool Temperature

One instrumented fuel element was put in the core for the experiments to evaluate the thermal hydraulic performance of the IPR-R1 Reactor [1]. The instrumented fuel is identical to standard fuel elements but it is equipped with three chromel-alumel thermocouples, embedded in the zirconium centerline pin. The sensitive tips of the thermocouples are located one at the center of the fuel section and the other two 25.4 mm above, and 25.4 mm below the center. Figure 4 shows the diagram and design of the instrumented fuel element [7]. The core upper view shown on the right in the Figure 1 we can see the instrumented fuel element in ring B of the core.
During the experiments it was observed that the temperature difference between fuel element and the pool water below the reactor core (primary loop inlet temperature) do not change for the same power value as can been seen on Fig. 5. With the instrumented fuel element in the hottest fuel element of the core (position B1), the power measured in linear channel (with the values corrected by the calibration results) was plotted as a function of the temperature difference between the fuel and the primary loop inlet temperature. The following polynomial expression was obtained that relates the two values:

\[ q = 2 \cdot 10^{-5} (\Delta T)^3 - 0.0045(\Delta T)^2 + 0.7666 \Delta T - 2.4475 \]  

(2)

![Graph](image.png)

**Figure 5. Fuel and inlet core water temperatures evolution.**
Where $q$ is the calibrated reactor thermal power, in [kW] and $\Delta T$ is the difference between the average fuel temperature and the primary loop inlet temperature, in [°C]. The determination coefficient obtained was one ($R^2 = 1$). Equation 2 was included in the data acquisition system and this new power measurement channel is available for the IPR-R1 TRIGA Reactor. After the experiments the instrumented fuel element was maintained in position B1 of the core to monitor the reactor power and core temperature in all reactor operation. Figure 6 shows reactor power measuring results using the linear neutron channel and the temperature difference channel method.

The fuel temperature limit defined in the IPR-R1 TRIGA Reactor Accidents Report [8] during steady state operation is 550 °C. A power operation limit over 1 MW was found based in this temperature and using Equation 2.

![Power Measurements](image)

**Figure 6.** Reactor power measured by neutron channel and by fuel element temperature.

Figure 7 shows one of the video-screens displays of the digital monitoring system computer that consolidates information, in real time, of the reactor power status. This screen monitors the power measured by the neutronics channels and by the three new thermal channels [4].

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3. CONCLUSIONS

The knowledge of the reactor thermal power is very important for precise neutron flux and fuel element burnup calculations. The burnup is linearly dependent on the reactor thermal power and its accuracy is important to the determination of the mass of burned $^{235}$U, fission products, fuel element activity, decay heat power generation and radiotoxicity. The thermal balance method presented in this report is now the standard methodology used for the IPR-R1 TRIGA Reactor power calibration [8]. The uncertainty value obtained does not differ significantly from another thermal calibration processes described in technical literature [2]. The heat balance and fuel temperature methods are accurate, but impractical methods for monitoring the instantaneous reactor power level, particularly during transients. For transients the power is monitored by the nuclear detectors, which are calibrated by the thermal balance method. On the other hand, the response of one nuclear detector is sensitive to the changes in the core configuration, mainly to the control rod position. This is important in research reactors, which do not have distributed absorbers for reactivity control and the normal mode of maintaining criticality is by insertion of control rods. The heating of the thermocouple due the gamma ray is negligible because the small mass and good thermal radiation equilibrium with the surrounding fuel.
ACKNOWLEDGMENTS

The authors thank to the operation team of the TRIGA IPR-R1 Reactor for their help and the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the financial support.

REFERENCES

POWER MEASURE CHANNELS OF THE IPR-R1 TRIGA RESEARCH NUCLEAR REACTOR BY THERMAL METHODS

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Abstract. The IPR-R1 Research Nuclear Reactor at the Nuclear Technology Development Centre – CDTN, in Belo Horizonte, is a TRIGA Mark I type reactor. The IPR-R1 is a pool reactor, and the fuel elements at the core are cooled by water natural convection. The heat removal capacity of this process is great enough for safety reasons at the current maximum 250 kW power levels of the reactor. However, a heat removal system is provided for removing heat from the reactor pool water. The water is pumped through a heat exchanger, where heat is transferred from the primary to the secondary loop. Power monitoring of nuclear reactors is always done by means of neutronics instruments (neutron flux measurement). In the IPR-R1 the power is measured by four nuclear channels. This work presents the results and methodology for the monitoring the power of this reactor by thermal processes. An improved method for thermal measuring channel is described using the fuel element and the water pool temperatures. Another thermal measuring channel consisted in the steady-state energy balance of the primary and secondary cooling loops of the reactor. For this balance, the inlet and outlet temperatures and the water flow of the cooling loops are measured.

Keywords: nuclear fuel element, nuclear reactor, thermal power, TRIGA reactor, instrumented fuel element

1. INTRODUCTION

The 250 kW IPR-R1 TRIGA Reactor (Training, Research, Isotopes, General Atomic), of the Nuclear Technology Development Center (CDTN), is a pool type reactor cooled by natural circulation of light water. The reactor fuel is an alloy of zirconium hydride and uranium enriched at 20% in 235U. Figures 1, 2 and 3 show two photographs and two drawings of the reactor pool and core. The reactor core has 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements. One of these steel-clad fuel elements is instrumented with three chromel/alumel thermocouples along its centerline. This instrumented fuel element was put in the reactor core in order to evaluate the thermal hydraulic performance of the IPR-R1 reactor under steady-state condition (Mesquita, 2005). Fuel temperatures were measured in various locations throughout the core with the use of the instrumented fuel element at different power levels. Three new processes for reactor power measurement by thermal ways were developed as a result of the experiments. These process make it possible on-line or off-line evaluation of the reactor power and the analysis of its behavior.

![Figure 1. Top view of the pool and core of the IPR-R1 TRIGA Research Nuclear Reactor](image-url)
Figure 2: The pool of the IPR-R1 TRIGA Nuclear Reactor

Figure 3: The core of the IPR-R1 TRIGA Nuclear Reactor
2. THE POWER MEASURING CHANNELS USING THE NEUTRONIC METHODS

Power monitoring of nuclear reactors is always done by means of nuclear detectors, which are calibrated by thermal methods. In the IPR-R1 Reactor four neutron-sensitive chambers are mounted around the reactor core for flux measurement. The type of chamber used and its position with respect to the core determine the range of neutron flux measured, as described below:

- The departure channel consists of a fission counter with a pulse amplifier that feeds a logarithmic count rate circuit and gives useful power indication from the neutron source level to a few watts.
- The logarithmic channel consists of a compensated ion chamber feeding a logarithmic (log n) amplifier and recorder and a period amplifier, which gives a logarithmic power indication on a recorder from less than 0.1 W to full power.
- The linear channel consists of a compensated ion chamber feeding a sensitive amplifier and recorder with a range switch, which gives accurate power information from source level to full power on a linear recorder.
- The percent channel consists of an uncompensated ion chamber feeding a power level monitor circuit and meter, which is calibrated in percentage of full power.

The nuclear instrumentation is used to detect neutrons when sub-critical multiplication occurs during the reactor start-up, and after achieving the criticality the variation of neutron flux, to obtain the automatic control of reactivity for maintaining a stable power level.

Unfortunately, the ionization chamber neutron detector measures the flux of neutrons thermalized in the vicinity of the detector. This signal is not always proportional to the integral neutron flux in the core and consequently to the core power. Besides the response of a single nuclear detector is sensitive to the changes in the core configuration, particularly to the control rod position. This is important in the TRIGA reactor, which do not have distributed absorbers for reactivity control and maintaining criticality is by insertion of control rods (Zagar et al., 1999).

3. THE POWER MEASURING CHANNEL USING THERMAL PROCESS

3.1. Power measuring by the thermal balance

The reactor core is cooled by natural convection flow of demineralized water in the reactor pool. Heat is removed from the reactor pool and released into the atmosphere by primary cooling loop, secondary cooling and a cooling tower (Fig. 4). Pool temperature depends on reactor power, as well as external temperature, because the latter affects heat dissipation in the cooling tower. The total dissipated power is determined by making the thermal balance of the inlet and outlet cooling water that flows through in the primary and secondary loops and the calculation of the heat losses. These losses represent a very small fraction of the total power (about 1.5% of total) (Mesquita et al., 2005).

The inlet and outlet temperatures are measured by four platinum resistance thermometers (PT-100) positioned at the inlet and at the outlet pipes of the primary and secondary cooling loops. The flow-rate in the primary is measured by the differential pressure on a orifice plate and a differential pressure transmitter, in the secondary the flow is measured by a flowmeter. The pressure transmitter was calibrated and an adjusted equation was obtained and added to the data acquisition system. The temperature measuring lines were calibrated as a whole, including thermometers, cables, data acquisition cards and computer. The adjusted equations were also added to the data acquisition system (Mesquita and Rezende, 2004).

The power dissipated at the cooling loop will be closer to the reactor power the closer the water temperature in the reactor pool is to the environment temperature. The steady state is reached after some hours of reactor operation, then the power dissipated in the cooling system added with the losses is equal the core power. The uncertainty in the power measurement considered all the uncertainty propagation from primary parameters, according to the methodology described by Coleman and Steele (1999). The uncertainty is calculated only for the power of the primary loop because it is now the standard power measuring system for the IPR-R1 Reactor (CDTN/CNEN, 2007).

The thermal power dissipated in the primary and secondary loops were obtained through a thermal balance given by the following equation:

\[ q_{\text{cool}} = \dot{m} \cdot c_p \cdot \Delta T \]  

(1)

Where \( q_{\text{cool}} \) is the thermal power dissipated in each loop [kW], \( \dot{m} \) is the flow rate of the coolant water in the loop [kg/s], \( c_p \) is the specific heat of the coolant [kJ/kg°C], and \( \Delta T \) is the difference between the temperatures at the inlet and the outlet of the loop [°C].

The data acquisition computer program calculates the power dissipated in the cooling loop with the collected data being used in Equation (1), and with the \( \dot{m} \) and \( c_p \) values corrected as function of the temperature of the coolant (Miller, 1989).
Figure 4. The reactor cooling system

To calculated the heat losses, one resistance thermometer (PT-100) were positioned inside the pool to measure the water pool temperature. A type K thermocouple was placed just above the pool surface to measure the air temperature at the reactor room. Two type K thermocouples were distributed around the pool, in holes in the reactor room floor, to measure the soil temperature. The core of the TRIGA Mark I IPR-R1 Nuclear Reactor is placed below the room floor, in the bottom of a cylindrical pool, 6.625 m deep and 1.92 m in diameter, whose upper surface is 25 cm below the level of the floor. The reactor pool transfers heat to the environment by conduction to the soil, through the lateral walls and through the bottom of the pool, and by convection and evaporation to the air at the reactor room, through the upper surface. All this losses is calculated by the data acquisition system as described by Mesquita et al. (2005).

Figure 5 shows the power evolution in the primary and secondary loops during one reactor operation. Table 1 presents the results of the thermal balance in this operation and some experimental data.

<table>
<thead>
<tr>
<th>Table 1. IPR-R1 TRIGA Reactor thermal balance.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average primary loop coolant flow rate</td>
</tr>
<tr>
<td>Average primary loop coolant inlet temperature</td>
</tr>
<tr>
<td>Average primary loop coolant outlet temperature</td>
</tr>
<tr>
<td>Power dissipated in the primary loop</td>
</tr>
<tr>
<td>Thermal losses from the reactor pool</td>
</tr>
<tr>
<td><strong>Reactor thermal power</strong></td>
</tr>
<tr>
<td>Standard deviation</td>
</tr>
<tr>
<td>Uncertainty in the measure of the reactor thermal power</td>
</tr>
<tr>
<td>Power dissipated in the secondary loop</td>
</tr>
</tbody>
</table>
3.2 Power measuring by fuel and pool temperature

One instrumented fuel element was put in the core for the experiments to evaluate the thermal hydraulic performance of the IPR-R1 Reactor (Mesquita, 2005). The instrumented fuel is identical to standard fuel elements but it is equipped with three chromel-alumel thermocouples, embedded in the zirconium centerline pin. The sensitive tips of the thermocouples are located one at the center of the fuel section and the other two 25.4 mm above, and 25.4 mm below the center. Figure 6 shows: a) The instrumented fuel element before it is put in the core; and, b) the core upper view with the instrumented fuel element in ring B. Table 2 presents some information about the instrumented fuel element (Gulf General Atomic, 1972). Figure 7 shows the diagram and design of this fuel element.

Figure 6. a) The instrumented fuel element. b) Core upper view with the instrumented fuel element in ring B
Table 2. Instrumented fuel element data.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heated length</td>
<td>38.1 cm</td>
</tr>
<tr>
<td>External diameter</td>
<td>3.76 cm</td>
</tr>
<tr>
<td>External fuel element active area</td>
<td>450.05 cm$^2$</td>
</tr>
<tr>
<td>External fuel area (U-ZrH$_1$)</td>
<td>434.49 cm$^2$</td>
</tr>
<tr>
<td>Fuel element active volume</td>
<td>423.05 cm$^3$</td>
</tr>
<tr>
<td>Fuel volume (U-ZrH$_1$)</td>
<td>394.30 cm$^3$</td>
</tr>
<tr>
<td>Power (total in the core = 265 kW)</td>
<td>4.518 kW</td>
</tr>
</tbody>
</table>

Figure 7. Diagram of the instrumented fuel element
During the experiments it was observed that the temperature difference between fuel element and the pool water bellow the reactor core (primary loop inlet temperature) do not change for the same power value as can been seen on Figure 8. With the instrumented fuel element in position B1 of the core (hottest fuel element), the power measured in linear channel (with the values corrected by the calibration results) was plotted as a function of the temperature difference between the fuel and the primary loop inlet temperature. The following polynomial expression was obtained that relates the two values:

\[ q = 2 \times 10^{-5} (\Delta T)^3 - 0.0045 (\Delta T)^2 + 0.7666 \Delta T - 2.4475 \]  

\[ (2) \]

![Figure 8. Fuel and inlet core water temperatures evolution](image)

Where \( q \) is the calibrated reactor thermal power, in [kW] and \( \Delta T \) is the difference between the average fuel temperature and the primary loop inlet temperature, in [°C].

The determination coefficient obtained was one \( (R^2 = 1) \). The Equation (2) was included in the data acquisition system and this new power measurement channel is available for the IPR-R1 TRIGA Reactor. After the experiments the instrumented fuel element was maintained in position B1 of the core to monitor the reactor power and core temperature in all reactor operation. Figure 9 shows reactor power measuring results using the linear neutron channel and the temperature difference channel method. It can be seen a delay in the second channel response due to the system thermal inertia.

The fuel temperature limit defined in the IPR-R1 TRIGA Reactor Accidents Report (CDTN/CNEN, 2007) during steady state operation is 550 °C. A power operation limit over 1 MW was found based in this temperature and using Equation (2).
Figure 9. Reactor power measured by neutron channel and by fuel element temperature

Figure 10 shows one of the video-screens displays of the digital monitoring system computer that consolidates information, in real time, of the reactor power status. This screen monitors the power measured by the neutronics channels and by the three new thermal channels (Mesquita and Rezende, 2004).

Figure 10. Power monitoring on the screen of the data acquisition system
4. CONCLUSION

The knowledge of the reactor thermal power is very important for precise neutron flux and fuel element burnup calculations. The burnup is linearly dependent on the reactor thermal power and its accuracy is important to the determination of the mass of burned 235U, fission products, fuel element activity, decay heat power generation and radiotoxicity. The thermal balance method presented in this report is now the standard methodology used for the IPR-R1 TRIGA Reactor power calibration (CTTN/CNEN, 2007). The uncertainty value obtained does not differ significantly from another thermal calibration processes described in technical literature (Zagac et al., 1999).

The heat balance and fuel temperature methods are accurate, but impractical methods for monitoring the instantaneous reactor power level, particularly during transients. For transients the power is monitored by the nuclear detectors, which are calibrated by the thermal balance method. On the other hand, the response of one nuclear detector is sensitive to the changes in the core configuration, mainly to the control rod position. This is important in research reactors, which do not have distributed absorbers for reactivity control and the normal mode of maintaining criticality is by insertion of control rods. The heating of the thermocouple due the gamma ray is negligible because the small mass and good thermal radiation equilibrium with the surrounding fuel.

5. ACKNOWLEDGEMENTS

The authors gratefully thank to the operation team of the TRIGA IPR-R1 Reactor for their help during the experiments and the Fundação de Amparo à Pesquisa do Estado de Minas Gerais (FAPEMIG) for the financial support.

6. REFERENCES


7. RESPONSIBILITY NOTICE

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SISTEMA DE MONITORAÇÃO EM TEMPO REAL DAS VARIÁVEIS DE OPERAÇÃO DO REATOR NUCLEAR DE PESQUISA TRIGA IPR-R1

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RESUMO

Os operadores de reator nuclear necessitam saber o comportamento do reator para compreendê-lo e operá-lo com segurança. Nos últimos dois anos, todos os parâmetros operacionais do reator de pesquisa TRIGA IPR-R1 foram monitorados e indicados em tempo real pelo sistema de aquisição de dados desenvolvido para o reator. Toda a informação foi armazenada em um disco rígido, no computador do sistema de coleta, formando uma base de dados, dos quais se obtém os operadores de reator e os parâmetros de segurança disponíveis para consulta em ordem cronológica. O programa de aquisição de dados foi atualizado e novos parâmetros operacionais do reator foram incluídos para aumentar as possibilidades de investigação e de experiência. Os registros das variáveis de operação do reator são importantes para as medidas de monitoramento e auditáveis de segurança, mostrar as tendências a curto e a longo prazo e para relatar as operações do reator à organização e às autoridades competentes. Este programa de aquisição de dados responde às recomendações da Agência Internacional de Energia Atômica - AIEA.

1. INTRODUÇÃO

O TRIGA IPR-R1 é um reator nuclear de pesquisa do tipo piscina, com núcleo refrigerado por circulação natural da água do seu poço. Localiza-se em Belo Horizonte, no Centro de Desenvolvimento da Tecnologia Nuclear, CDTN. Opera atualmente com potência térmica máxima de 100 kW, e seu núcleo está configurado para operar em potências de até 250 kW, em fase de licenciamento. Possui um sistema de refrigeração forçada dotado de circuito primário e secundário, que refrigera a água do poço e diminui os níveis de radiação na saída do reator durante operações com potência acima de 50 kW. Conta com um Sistema de Aquisição e Tratamento de Dados, desenvolvido no ambiente VXiDAQ da Advantech CO® [1], implantado em 2004, no qual todos os seus parâmetros de controle são visualizados em tempo real e armazenados em disco rígido.

Com a implantação do sistema de coleta de dados, os dados coletados são armazenados em arquivos tipo “.txt” – bloco de notas – organizados em pastas, cronologicamente. Para a
utilização dos dados para construção de tabelas e realização de análises é necessário migrar esses dados para aplicativo adequado, neste caso é utilizado o editor de planilhas Microsoft Excel®. Este trabalho trata da inclusão de parâmetros no monitoramento em tempo real e da criação de um banco de dados dos parâmetros de operação do reator TRIGA IPR R1, de maneira a facilitar o acesso aos dados e sua utilização pelos pesquisadores dessa instituição.

Os registros desses dados, e posterior arquivamento, são importantes para análises imediatas ou subsequentes de segurança, para mostrar o comportamento do reator e suas tendências a curto e a longo prazo, e para relatar as operações à instituição e às autoridades externas. O programa de aquisição de dados responde às recomendações da Agência Internacional da Energia Atômica – IAEA [2].

2. MÉTODOS

O Sistema de Aquisição de Dados foi desenvolvido para acompanhar o desenvolvimento dos parâmetros de controle do reator e para subsidiar posteriores estudos sobre a evolução dos mesmos, além de possibilitar a realização de registro durante operações experimentais [3].

O sistema coleta os sinais da mesa de controle do reator, do bastidor de instrumentação, e de termopares, por meio de duas placas condicionadoras que direcionam os sinais a uma placa conversora analógico/digital modelo PCI-818hd fornecida pela Advantech Co. Ltd [1] instalada num computador. Estes sinais, depois de captados pela placa conversora A/D, são processados pelo programa de aquisição de dados VisiDAQ, também da Advantech Co. Ltd [1], com linguagem de programação Microsoft Visual Basic for Applications, através de equações de ajuste obtidas por processos de calibração para que, nas interfaces gráficas desenvolvidas, fossem mostrados os valores reais de cada parâmetro operacional do reator, bem como suas evoluções em gráficos específicos.

2.1 Interfaces Gráficas

Foram incluídos no sistema de aquisição de dados novas variáveis: nível de água do poço e perda de reatividade por envenecimento por xenónio e por efeito temperatura. Desde a primeira versão, o programa conta com cinco interfaces visuais, nas quais os parâmetros coletados são dispostos em grupos para a facilitar a compreensão imediata e acompanhamento em tempo real das variáveis do reator. As interfaces são:

- Partida: contém informações importantes para uma partida segura do reator, como posição de barras de controle, taxa de contágios do canal de partida, período, reatividade, perda de reatividade por envenenamento por xenónio e por efeito temperatura [Fig. 1].
- Canais de Potência: informa todas as leituras de potências realizadas no reator. Além disso, possui o botão GRAVA que deve ser acionado para que os dados lidos passem a ser registrados, bem como possui um relógio que mostra o tempo de operação em segundos, minutos e horas [Fig. 2].
- Níveis de radiação: mostra as cinco medidas relativas à radioproteção no ambiente do reator, aerossóis, poço, área, entrada e saída do circuito primário de refrigeração.
- Potências: outra interface que também mostra os valores das potências do reator, porém em um gráfico. Apresenta a temperatura média do combustível instrumentado e a temperatura da água do poço do reator. Possui um relógio que informa o tempo de operação em horas.
Figura 1. Interface Gráfica "Partida" do Sistema de Aquisição de Dados do Reator TRIGA IPR R1.

Figura 2. Interface Gráfica "Canais de Potência" do Sistema de Aquisição de Dados do Reator TRIGA IPR R1.
• Refrigeração e Temperatura: possibilita a visualização de parâmetros relacionados à termohidráulica do reator. Mostra todas as medidas de temperaturas realizadas no reator, bem como a temperatura do combustível instrumentado. O combustível instrumentado possui três termopares em seu eixo central e encontra-se posicionado no local mais quente do núcleo. A temperatura deste combustível é a única medida que não é mostrada na mesa de controle do reator. O registro desta temperatura pode ser efetuado mesmo com a mesa de controle desligada. Esta interface mostra também a vazão e o nível de água do poço, possuindo também um contador de tempo [Fig. 3].

![Figura 3. Interface Gráfica "Refrigeração e Temperaturas" do Sistema de Aquisição de Dados do Reator TRIGA IPR R1.](image)

2.2 Construção do Ambiente Gráfico

A construção das interfaces gráficas é feita no próprio programa em uma tela especial denominada "Task" [Fig. 4]. Foram inseridas funções utilizando a barra de ferramentas do programa, para a adição dos novos parâmetros coletados: nível de água do poço, perda de reatividade por envenenamento por xenônio e por efeito temperatura.

A medida do nível de água do poço (N), é retirada da mesa de controle do reator com uma ligação em paralelo com o sinal de entrada da mesa (AI23) e é ajustada pela equação abaixo:

\[
N = (AI23 \times 19.571) + 0.663
\]  

(1)
Estes ajustes são realizados para que, na interface Refrigeração e Temperaturas, seja mostrado o mesmo valor que aparece no display da mesa de controle.

2.3 Armazenamento de Dados

Após a operação, os dados coletados são gravados em quatro arquivos tipo “.txt”. Cada um destes arquivos contém um grupo de parâmetros. Os dados são coletados no computador do sistema de coleta de dados e compilados para outro, salvos em pastas, em ordem cronológica e cada pasta corresponde a um dia de coleta. Os dados das coletas salvos em formato “.txt” são copiados e tratados na planilha eletrônica, e as informações passam a ser salvas em um único arquivo tipo “.xls”, ou seja, cada arquivo formato “.xls” também corresponde a um dia de coleta. Depois de tratados, são construídos gráficos na planilha eletrônica Excel®.

3. RESULTADOS

Desde 2004 o Sistema de Aquisição de Dados começou a ser utilizado frequentemente no reator [4]. Atualmente a coleta de dados já faz parte da rotina de operações o que contribui para manter o banco de dados sempre atualizado [5]. Os gráficos construídos a partir das coletas permitem a realização de estudos comparativos entre dados teóricos e experimentais, e também uma visualização ampla do comportamento dos parâmetros do reator, permitindo intervenções precoces para a prevenção de acidentes. A nova versão do sistema de aquisição de dados do reator TRIGA IPR R1 conta com o registro e análise “on line” de mais variáveis. O banco de dados formado pelos registros digitais das operações do reator continua no mesmo formato, ou seja, organizando cronologicamente as coletas em arquivos “.txt” e “.xls”. Com o registro digital e acompanhamento em tempo real dos parâmetros do reator aumentou-
se a velocidade de transmissão e de acesso às informações e a capacidade de produzir, armazenar e transmitir informações sobre o reator.

4. CONCLUSÃO

A criação do banco de dados com todas as coletas realizadas no reator é de suma importância, pois possibilita a execução de pesquisas e o desenvolvimento de trabalhos e métodos a partir da análise dos parâmetros de operação do reator. Espera-se que, com banco de dados desenvolvido e a partir das análises decorrentes, aumente-se o nível de utilização do reator TRIGA em pesquisa e desenvolvimento científico, em tecnologia e produtos na área nuclear, e que se colabore no aumento da segurança e confiabilidade na operação do Reator TRIGA IPR-R1. Observa-se que o sistema de controle deve ser ampliado e modernizado a fim de se aumentar a potência do reator com segurança, permitindo mais possibilidades de utilização do reator para a pesquisa em novas áreas, análise de materiais como obras de arte, gemas, semicondutores e matrizes orgânicas e biológicas, alimentos, fármacos e culturas de células, além de possibilitar a utilização de outras técnicas, como a neutrôgrafia e a difração de nêutrons, aplicadas também ao estudo e a caracterização de materiais.

A maior demanda por informações do reator decorre do interesse dos usuários e da qualidade das informações disponíveis, bem como das facilidades de acesso. Atualmente, os registros digitais são gerenciados por um operador do reator, de modo que, vistos em conjunto, assemblham-se muito mais a um sistema de arquivos tradicional do que a um banco de dados institucional e integrado. Pode-se facilitar o acesso aos dados e reduzir desvantagens características do atual sistema. Entre as desvantagens, pode-se mencionar o elevado tempo de consulta, porque para se ter acesso a um parâmetro é necessário abrir um arquivo que contém todos os parâmetros e selecionar o desejado, e esses arquivos são grandes. É necessária a redundância dos dados, entendida como a possibilidade de a mesma informação estar contida em mais de um arquivo (backup). Os registros são armazenados em dois computadores, o que envolve maior trabalho de atualização, isto é importante para garantir que, caso um computador apresente defeito, os dados não sejam perdidos.

REFERÊNCIAS

ABSTRACT

Nuclear reactor operators need to know the basic behaviour of reactors in order to understand and safely operate them. In the last two years, all operational parameters of the IPR-R1 TRIGA Reactor were monitored and displayed on-line by using a data acquisition system developed for the reactor, and to show real-time performance of this plant. All information was stored in a computer hard disk with an accessible historical database in order to make available chronological information about the reactor performance and behaviour. This data acquisition program was updated, new reactor operational parameters were included in order to increase the possibilities of investigation and experiments. Records of process variables of the reactor are important for immediate or subsequent safety analyses, to show the short and long term trends, and for reporting the reactor operations to the organization and external authorities. The data acquisition program responds to recommendations of the International Atomic Energy Agency - IAEA.
THE BRAZILIAN IAEA EXPERT MISSIONS IN THE RECOMMISSIONING OF THE IAN-R1 TRIGA REACTOR

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ABSTRACT

This paper describes the works that were done by two Brazilian researchers of CDTN/CNEN, during the reactivation of the IAN-R1 Reactor in Bogotá (Colombia) in October 2005. The main aim of this mission was to participate in the Ad-hoc committee, established by Colombian Authorities, as International Atomic Energy Agency experts, to follow the recommissioning of the reactor IAN-R1, and assist the Colombian staff in the safe operation of the reactor. The work was carried out during two weeks and consisted in reviewing the operational procedures, results and records and providing lectures for the operating group. The reactor core was brought critical by adding two clusters (8 fuel elements); the three control rods were calibrated; the excess reactivity and shutdown margin were determined; and the thermal power evaluation was performed.

1. INTRODUCTION

Two researchers from the Nuclear Technology Development Center (CDTN), worked as experts of the International Atomic Energy Agency (IAEA), in the activities of hot commissioning of IAN-R1 Research Reactor [1] (Fig. 1), in Bogotá. The operation of this reactor, the only one in this country, started the recovery of the Colombian nuclear program.

Figure 1. The IAN-R1 TRIGA Reactor and console.

The reactor was designed in 1965 as a small, 10 kW facility using aluminum plate-type, highly enriched fuel (MTR), under the United State’s Atoms for Peace Program. General
Atomic has been involved in a gradual upgrading of the facility since 1988, and, in late 1994, a tripartite contract was signed with the IAEA, Colombian Authority and General Atomic to manufacture, install and commission the reactor with TRIGA-type, conversion to TRIGA low-enriched fuel [2, 3], and increasing the power of this reactor to 100 kW.

The nuclear activities in Colombia were interrupted in 1998 with the extinction of the Institute of Nuclear Affairs (IAN). In this year, eight fuels elements were removed from the reactor core to kept IAN-R1 TRIGA Reactor in a secure subcritical condition. It was brought into an extended shutdown condition in 1998 after the core conversion, which included the commissioning conducted by the supplier (General Atomic) [4]. The Colombian Authorities decided to reactivate the nuclear activities in the country with IAEA support [5]. In 2005, the authors of this paper were invited to participate in this work, due to their experience in working at experimental reactor physics and thermal hydraulic of the CDTN’s IPR-R1 TRIGA Reactor.

The reactor IAN-R1 is a swimming pool type with concrete shield and two beamports. The fuel (U-ZrH$_{1.6}$) is contained in 4-rod clusters. The core configuration is a rectangular grid plate that holds a combination of 4-rod and 3-rod clusters. The 3-rod clusters provide a fourth cluster space to be used either for in-core irradiation or control rod locations. The core contains 50 fuel rods, 3 control rods and 3 in-core water filled experimental locations. The maximum core power level is 100 kW corresponding to a thermal neutron flux level varying from $1.9 \times 10^{12}$ to $4.2 \times 10^{12}$ n/cm$^2$.s, depending on the core locations [6]. The assembly is located inside an open tank full of light water which acts as biological shielding, partial neutron moderation and core coolant. The reactor core is cooled by natural circulation. The tank water is cooled by the primary and secondary systems.

2. DESCRIPTION OF THE ACTIVITIES

The two clusters (8 fuels rods) removed from the core in 1998 were replaced. The reactor was critical at 100 W during one hour. The Ad-Hoc committee was present during these experiments. When the reactor was critical, it could be observed that the reactivity values of each control rod are almost the same. The top view of the core without the two clusters and the core configuration after the fuel loading are shown in Fig.2.

![Figure 2. IAN-R1 reactor core and the configuration after the fuel loading.](image-url)
2.1 Control Rod Calibration

All three-control rods were calibrated by the positive period method. The method consists of withdrawing the control rod from a known critical position through a small distance. Each successive step is compensated by lowering the other control rod just enough to reestablish criticality. In this process the control rod under calibration proceeds from the most inserted position (maintaining the reactor critical) to fully removed. The reactor period was obtained using the doubling time (DT), that is the time required for the power to increase by a factor of two.

The doubling time was measured with two digital chronometers, observing the power showed in the digital display in the console, 2 minutes after withdrawing the control rod under calibration, in order to finish the transition region. The reactivity associated with the measurement was gotten from the graphical form of the Inhour equation. It is important to note that for periods longer than one second, the curve is essentially independent of both $\ell$ and $\beta$. The reactivity measurements were performed at a low power so that the temperature increase during the experiment was negligible.

The Shim 1 and Shim 2 rods were intercalibrated. The idea was to measure one control rod in presence of another rod, used for compensating the reactivity introduced by step withdrawal of the measure rod. The Regulating, Shim 1 and Shim 2 rods worth were 3.25 $, 2.89 $ and 3.39 $, respectively. The three control rods have sufficient reactivity worth to shutdown the reactor, independently.

During these calibrations one power measuring channel could not be used because the fission counter associated to it was not functioning (water had entered into the detector). We decided to continue the tests since the reactor was operating at low power, and the other two power measuring channels were still working. The initial operation checklist was accomplished and it was observed that all protection devices were available. Besides the manual scram button on the operator's control panel and close to the reactor pool, there are many automatic shutdown circuits (scram circuits).

2.2 Core Excess Reactivity and Shutdown Margin

The excess reactivity ($\rho_{ex}$) of the core was determined from different control rods critical positions, at low power, and the correspondent calibration curves. The average excess reactivity value obtained was $(2.18 \pm 0.08) $.

The total reactivity worth of the control system was 9.53 $. With a core excess reactivity of 2.18 $, the shutdown margin with the most reactive rod (Shim 2) stuck out of the core was 3.96 $. The shutdown margin is that margin the reactor can be shutdown from a critical condition, and is given by the difference between the reactivity worth of the considered control rods (the most worthy rod is assumed fully withdrawn) and the core excess reactivity. Table 1 presents the values of the control rods worth, the core excess reactivity, and the shutdown margin for IAN-R1 core configuration.
Table 1. Results of reactivity

<table>
<thead>
<tr>
<th>Parameter</th>
<th>$\rho$</th>
</tr>
</thead>
<tbody>
<tr>
<td>REGULATING Worth</td>
<td>3.25</td>
</tr>
<tr>
<td>SHIM 1 Worth</td>
<td>2.89</td>
</tr>
<tr>
<td>SHIM 2 Worth</td>
<td>3.39</td>
</tr>
<tr>
<td>Excess Reactivity</td>
<td>2.18</td>
</tr>
<tr>
<td>Shutdown Margin – SHIM 2 Out</td>
<td>3.96</td>
</tr>
</tbody>
</table>

2.3 Thermal Power Estimate by Calorimetric Procedure

Before starting this test the fission detector had already been repaired, and it was operating properly. It was recommended that for routine operations the two fission counters and the ion chamber should be available. The thermal power calibration was performed using the calorimetric method [7]. Some thermocouples were put along the pool, and the top of the pool was thermally isolated. The reactor stayed critical at a constant power of 8 kW, indicated in the console, during about 3 hours, with manual power corrections because the automatic control system failed. The control rod initial positions were: Shim 1 (655), Shim 2 (661) and Regulating (674). The primary cooling system was switched off, and the rate of temperature rise was determined. With the specific heat of the system and water volume of the pool, the core power was then determined from the measured rate of temperature rise from operation of the reactor. During the experiment, all the pool temperatures were collected in intervals of 30 minutes. Fig. 3 shows the positions of the thermocouples in the pool, and the average water temperature versus the running time, during the thermal power estimation.

![Figure 3. Pool water temperature increase during the thermal power estimation.](image-url)
At 8 kW in the control console the radiation level was approximately of the same order of that one obtained during General Atomics tests [4]. The Cerenkov radiation could already be visualized in the reactor core (Fig. 4). There was a scram for high radiation, indicating that the actual power was larger than the console indication. The power obtained by the calorimetric method was 30 kW with an uncertainty of 20%.

![Figure 4. The Cerenkov radiation during the thermal experiment.](image)

In the next day, the reactor was turned on in critical conditions with the control console indicating 8 kW. The positions of the ion chamber, and the two fission counters were adjusted until the console indications became 30 kW.

3. CONCLUSIONS

The neutronic and thermal-hydraulic parameters obtained in the recommissioning program were close to those found by General Atomics in the tests conducted in 1997 [4].

The reactor core was brought critical by adding two clusters (8 fuel elements). The control rods were calibrated by the positive period method, and the Regulating, Shim 1 and Shim 2 control rods worth were 3.25 $, 2.89 $ and 3.39 $, respectively. The three control rods worth were almost the same, and they have sufficient reactivity to shutdown the reactor, independently. The excess reactivity obtained for the proposed core was 2.18 $, and the shutdown margin, with the most reactive rod stuck out of the core, was 3.96 $, hence greater than the minimum safety limit required. The thermal power calibration was performed using the calorimetric method. It was determined that the real reactor power was 30 kW with an uncertainty of 20%.

The IAN-R1 Reactor installation has good computer equipment and electronic instrumentation, a data acquisition system has already been implemented [8], as well the communication and physical protection systems [9]. The control bar graph display of the
console screen is very friendly. There is a No-Break that feeds the control console and the control rods electromagnets. So, in the event of electrical energy failure the operator has time to make decisions.

The recommissioning of the IAN-R1 TRIGA research reactor was successfully completed, in safety conditions. The technological interchange and cooperation among the American Latin countries were a very positive fact of this mission.

ACKNOWLEDGMENTS

The authors would like to thank the operation staff of the IAN-R1 Research Reactor, and to compliment IAEA, and, specially, Dr. Heriberto José Boado Magan for giving incentive to the cooperation among Latin American countries in the solution of their nuclear problems.

REFERENCES

The Operational Parameter Electronic Database of the IPR-R1
TRIGA Research Reactor

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ABSTRACT

Nuclear reactor operators need to know the basic behaviour of reactors in order to understand and safely operate them. In the last four years, the main operational parameters of the IPR-R1 TRIGA Mark I Reactor of Nuclear Technology Development Center (CDTN) at Belo Horizonte, Brazil, have been monitored and displayed on-line by using a data acquisition system developed for this reactor. Besides showing the real-time performance of the plant, the system stores the information in a computer hard disk, with an accessible historical database, in order to make the chronological information on reactor performance and its behaviour available to users. Some of the parameters stored are: the control rod positions and its reactivity, the reactor power, the fuel and water temperatures, the radiation levels, the primary cooling system flow, the water pool level and so on. Records of process variables of the reactor are important for immediate or subsequent safety analyses to show the short and long term trends, and to report the reactor operations to the organization and external authorities. This paper describes the data acquisition system, the developed electronic database and operational procedure used by the IPR-R1 TRIGA reactor staff to storage the operational parameter.

1. INTRODUCTION

The IPR-R1 TRIGA Research Reactor, located at the Nuclear Technology Development Center – CDTN, has been operating for 48 years. The operational parameters are monitored and displayed by analog meters located at reactor console. The reactor operators registered the most important operational parameters manually in a logbook. This process is quite useful, but it can involve some human errors, parallax error, time delay, etc. It is also impossible for the operators to take notes of all variables involving the process mainly during fast power transients. Due to the experiments on neutronics, thermohydraulics and reactor power calibrations, it was necessary the development of a data acquisition system to make possible the tests [1, 2, 3, 4]. Now, this system is part of the IPR-R1 reactor operational procedures, and it is used to register the operational parameters and to maintain the electronic files. The system responds to the International Atomic Energy Agency - IAEA recommendations on the monitoring and registration of the operational variables [5]. The Figure 1 shows the reactor cooling system diagram and the localization of the instrumentation used to monitor the operational parameter.
2. HARDWARE DESCRIPTION

The reactor instrumentation includes four neutron channels that are fed by three neutron-sensitive ion chambers (two compensated and one uncompensated), and by one fission chamber located around the core. The detector's signals are amplified in a voltage amplifier. The amplified output voltage is between 0 to \( \pm 10V \) which is the input sign of the Multiplexing Board and to the Analog to Digital Converter card of the PC with the associated software. The other analog signs, collected by the data acquisition system, are outputs of the reactor control console and from some digital indicators or directly from the thermocouples. The measure data are shown in the LCD computer video monitor. The main components of the instrumentation are described in the next sections.

3. DATA ACQUISITION CARDS

3.1. Amplifier and Multiplexing Cards

The analogical signs are received in three cards model PCLD-789 by Advantech Co [6] connected in cascade, each one with 16 channels which totalize 48 inputs. These cards prepare the signs amplifying and filtering the noises, and make the connection for a unique analogical output (multiplex action). One of the cards receives the signs directly from the thermocouples (range of \( \pm 100 \, \text{mV} \)). It has a sensor that measures the temperature and makes
the compensation of the cold junction adjusting the measured value. The other cards receive
the signs from the control console (range of ±10 V).

3.2. Analog to Digital Conversion Card

The outputs of the three conditioning cards are addressed to the analog input plug of the data
acquisition card, model PCL-818hd by Advantech Co [6]. This is a high-speed data
transference card installed in the computer case, which transforms the analog input signs into
digital sign.

4. DATA ACQUISITION SOFTWARE

The data acquisition system is programmed with a set of icons that represents controls and
functions, available in the menu of the software. Such a programming is called visual
programming. The user interface consists of two parts - a front panel and a diagram. The
front panel is used for input, output controls, and to display the data whereas the circuit
resides on the circuit board. The front panel has buttons, indicators and graphics and display
functions. This interface also calculates mean and standard deviation of the data and plots it.

The main indications of the control console are collected by the data acquisition system,
including the positions of the three control rods. These signs come from the neck of the
instruments and from the reactor control console and they input in channels 1 to 15 of the
Card 2. A description of all signs collected from the control console is not presented in this
paper. It was accomplished all the answers of the parameters collected and the equations used
to transform the signals of Volt into engineering units were introduced in the data acquisition
program. The program presents four screens, one of them is shown in Fig. 2 (Control, Start
up Channel, Period and Reactivity). The other screens are: Radiation Levels, Power Channels
and Cooling System and Temperatures.

4.1 Control, Startup Channel, Period and Reactivity

On the screen, shown in Fig. 2, the start up of the reactor can be accompanied through the
neutron evolution counting rate. The positions of the three control rods of the reactor can be
visualized in graphics and in digital indicators. The evolution of the control rods position and
its reactivity, in the last 60 minutes, are shown in three graphics. The reactivity of the reactor,
in percent and in milliunit, is given by digital counters. This screen also shows the loss of
reactivity as a function of power and time, the positive period of the reactor (T) in [s] and the
start up rate (SUR) in [rpm].

4.2. Radiation Levels

The gamma radiation levels at the reactor area are measured in the following positions: in the
Control Room (APR iso); in about 30 cm above the reactor pool (POÇO); at 2 m above
the reactor pool (AREA); at the inlet piping of the primary cooling loop heat exchanger
(ENTRADA PRIMARIO); in the hot exchange system (SÉMINAL), and at the outlet piping of
the secondary cooling loop heat exchanger (SAIDA SEUNARIO). These radiation levels of
the monitoring channels are shown on the screen in analog and digital indicators and their
evolution, in the last 60 minutes, are presented in graphic ways.
4.3. Power Channels

This screen shows the evolution of the reactor powers supplied by three conventional neutron channels of measurement: Logarithmic Channel, Linear Channel and Percent Power Channel. Digital indication and graphics show the last 60 minutes values. The evolution of the power dissipated in the primary and secondary cooling systems is also shown. After several hours of reactor operation, when it is reached the thermal balance with the environment, the power of the reactor will be closer to the power dissipated in the primary coolant loop, and the thermal losses will be smaller. Those losses value are indicated on the screen. The reactor power is monitored also by the increase of the temperature in the center of the instrumented fuel.

4.4. Cooling System and Temperatures

All the following parameters of the primary and secondary cooling loops are monitored and shown on the screen: the average values of the inlet and outlet temperatures of the primary and secondary loops, of the flow rate, and of the temperatures in the reactor pool, and their standard deviations; the power dissipated in the primary and secondary cooling loops; the air temperatures above the reactor pool and in two points of the soil; the average temperature taken from three thermocouples of the instrumented fuel element; and the time elapsed since the program has begun, in [s], [min] and [h].

![Figure 2. Control, Start up Channel, Period and Reactivity Screen of the IPR-R1 TRIGA Data Acquisition System](image)

DATABASE SYSTEM

The database is a computerized collection of the operational parameters, organized so that it can be expanded, manipulated, and retrieved rapidly for various uses. Within the electronic database, a computer program assists the user in selecting desired pieces of data. The data are recorded in five separated text-files that permit to register 40 parameters. In all files, the first
column is always the time registration in [s]. The data collection and recording frequency can be adjusted starting from 1.0 Hz to 1.0 kHz.

The Figure 3 shows the IPR-R1 database with all operation files from the year of 2002 to the last operation at this moment. The data is recorded in text files (.txt) and after it is passed to Excel software to allow the visualization of variables evolution (Fig. 4). Safety backup is made to other computers and also available in the Ethernet.

Figure 3 – IPR-R1 TRIGA Operation Database in Microsoft Word Files and Microsoft Excel Files

The trend of a series of data can be analyzed from the curve charts. An example of display pattern of results data are showed in Fig. 4.

Figure 4 – Example of the Data Display Pattern of Results Data
5. CONCLUSIONS

The new data acquisition based on the microcomputer has been designed and developed to allow the real-time collection of all IPR-R1 TRIGA Reactor operational parameters and information from the reactor console. They are fed into the computer through an interface unit that includes hardware like multiplexer, A/D converter and computer interface. Information on all aspects of reactor operation is displayed on the computer screen. The color graphic monitors can display real-time operation data in concise, accurate, and easily understood formats. Bar graph indicators, and visual and audible enunciators are also provided. Information displayed on the monitor is recorded on hard disk copy. The system collects data during reactor operations and stores it in a historical database. This record is a powerful tool that can be used for operations review and maintenance troubleshooting. In every operation of the research reactor about forty variables are registered by the data acquisition system. The system provides data for the last five years, (since 2004).

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4th WORLD TRIGA USERS CONFERENCE.
Thermal behaviour of the IPR-R1 TRIGA nuclear reactor

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Abstract: Experimental and analytical studies have been performed, at Nuclear Technology Development Centre - CDTN (Belo Horizonte), to find out the temperature distribution in the IPR-R1 TRIGA Research Nuclear Reactor as a function of power and position in the reactor core. The basic safety limit for the TRIGA reactor system is the fuel temperature, both in steady-state and pulsed mode operation. The time-dependence of temperature will not be considered here, hence only the steady-state temperature profile will be studied. The experimental results for fuel and coolant temperatures in the reactor core, at different reactor power levels, have been compared with theoretical data and some results from other TRIGA reactors.

Keywords: temperature; heat transfer; nuclear fuel rod; natural circulation; subcooled nucleate boiling; instrumented fuel element; TRIGA reactor.


Biographical notes: Dr. A. Z. Mesquita is graduated in Electrical Engineering (1978) and M.Sc. in Nuclear Engineering (1982) from Federal University of Minas Gerais (Brazil) and Ph.D. in Chemical Engineering from State University of Campinas (Brazil) (2003). He is Senior Operator of the IPR-R1 TRIGA Reactor of Belo Horizonte. Mr. H. C. Rezende is graduated in Mechanical Engineering (1979) and M.Sc. in Nuclear Engineering (1985) from Federal University of Minas Gerais (Brazil).

2. Introduction

The 250 kW IPR-R1 Reactor of the Nuclear Technology Development Center (CDTN) is cooled by natural circulation, like other TRIGA reactors. The reactor core (Fig. 1) has 59 aluminum-clad fuel elements and 5 stainless steel-clad fuel elements with 20 % enrichment and 8.5 wt % in uranium. One of these steel-clad fuel elements, shown in Fig. 2, is instrumented with three thermocouples positioned along its centraline. Experiments were made in order to evaluate the thermal hydraulic performance of the IPR-R1 reactor. The temperature distribution under steady-state condition was measured as a function of the reactor power and of the position in the reactor (Mesquita, 2005). Temperatures were measured in many locations throughout the reactor pool, in the fuel element centraline and in the coolant channels in the core.
Figure 1  Top view of the IPR-R1 TRIGA core

Figure 2  The instrumented fuel element
Data are obtained from the console and from a data acquisition system computer that was developed as part of this research project by Mesquita et al., (2004). Some of the parameters collected are power, fuel and channel temperatures, water flow, and control rod insertion position.

2. Temperature distribution in the core and in the pool

The original fuel element at the reactor core position B1 was removed and replaced by an instrumented fuel element with three type K thermocouples positioned along the fuel centreline (Fig. 1 and Fig. 2). Position B1 (Fig. 3) is the hottest location in the core, according to the neutronic calculation (Dalle, 2005). Two type K thermocouples were inserted into the core in two channels close to position B1 and measured the inlet and outlet temperatures in the hot channel. Nine thermocouples and one platinum resistance thermometer were used to monitoring the reactor pool temperature. The thermocouples were positioned in a vertical aluminium probe and the first thermocouple was 143 mm above the core top grid plate. The reactor thermal power calibration was carried out at 250 kW (Mesquita et al. 2005). After finding the power reference value, the instrumented fuel element was replaced to new positions in each one of the fuel rings from B to F. At the same way, the two coolant channel thermocouples were replaced to channels close to the instrumented fuel element. Experiments were carried out with the power changing from about 50 kW to 250 kW in 50 kW steps for each position of the instrumented fuel element.

Figure 2  Temperature measures places in the reactor core and the instrumented fuel element
Figure 3  Temperature measures places in the core top grid plate of the reactor core

Figure 4 shows the theoretical radial power profile [5] and the experimental fuel temperature and the inlet/outlet coolant temperatures in the channel that is the closest to the instrumented element, for the power of 265 kW. The experimental results found in the I.T.U. TRIGA Mark II Reactor at the Istanbul University (Ozkul and Durmayaz, 2000) were also plotted on the same graphic. Figure 5 shows the experimental profiles of inlet and outlet channel coolant temperature together with the same temperature curves calculated using the PANTERA code (Veloso, 2005). Results of fuel
Thermal behaviour of the IPR-R1 TRIGA nuclear reactor

temperature versus power experiments are shown in Fig. 6 for each core ring as a function of the reactor thermal power.
The experimental coolant exit temperature for each core ring is shown in Fig 7 as a function of the reactor power. The shape of these curves is similar to that predicted by the theoretical model.

Figure 4  Core temperature radial profile at 265 kW thermal power

Figure 5  Temperature radial profiles in the core channels at 265 kW thermal power
A. Z. Mesquita and H. C. Rezende

Figure 6  Fuel temperatures as function of the reactor power in all core rings

Figure 7  Outlet coolant temperatures as function of the thermal power
2.1. Temperature Profile in the Channel

The experimental temperature profile of the coolant water in Channel 1 is shown in Fig. 8 as a function of the axial position, for the powers of 265 kW and 106 kW. The shape of these curves is different from that predicted from the theoretical model (PANTERA Code) (Veloso, 2005). Ideally, the coolant temperature would increase along the entire length of the channel, because heat is being added to the water by all fuel region in the channel. Experimentally, the water temperature reaches a maximum near the middle length and then decreases along the remaining channel. Figure 8 shows also the experimental results for other TRIGA reactors (Bitke and Gazave, 2000); (Bürs and Vaurio, 1966) and (Flang, 1988).

Figure 8 Temperature profile along the Channel 1

2.2. Pool Temperature
The reactor operated during a period of about eight hours at a thermal power of 265 kW before the steady state was obtained. Figure 9 shows the water temperatures evolution at the reactor pool, and the inlet and outlet coolant temperature in the core's hottest channel, up to the begin of the steady state. The results showed that the thermocouples positioned 143 mm over the top grid plate (Inf 7) measure a temperature level higher than all the other thermocouples positioned over the reactor core. It means that the chimney effect is not much high, less than 400 mm above the reactor core, in agreement with similar experiments reported by Rao et al. (1988).

Figure 9  Temperatures patterns at 265 kW thermal power

In steady state operation at 265 kW the pool cooling system was turned off during about 15 min. Figure 10 shows the behaviour of fuel, channel outlet and pool average temperatures. The pool temperature begins to arise and the fuel and channel temperatures remain almost at the same values.

One type K thermocouple was put in Position 40 of the rotary specimen rack for about 2 hours with the IPR-R1 operating at the power of 100 kW. Figure 11 shows the results from the temperature data in the fuel element, reactor pool, reactor room and specimen rack. At this power the higher temperature in the irradiation facility was 41.5 °C.

Figure 10  Temperature behaviours after turning off the cooling system
3. Conclusions

The experiments confirmed the efficiency of the free circulation in removing the heat produced in the reactor core by nuclear fission. The instrumented fuel temperature stayed almost constant at about 300 °C at 265 kW thermal power, with the primary cooling loop circulation turned off. The data taken during the experiments provides an excellent picture of the thermal performance of the IPR-R1 Reactor. The experimental data also provides information, which allows the computation of other parameters, such as the flow velocity through the core and the heat transfer coefficient (Mesquita and Rezende, 2006). The theoretical temperatures and flow velocity were determined under ideal conditions (Velez, 2005). The actual coolant flow is quite different because of the inflow of water from the core sides (cooler than its centre). There is a considerable coolant cross
flow throughout the channels. The temperature measurements above the IPR-R1 core showed that water mixing occurs within the first few centimetres above the top of the core, resulting in an almost uniform water temperature. Finally, the temperature at the primary loop suction point at the pool bottom has been found as the lowest temperature in the reactor pool.

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The authors gratefully thank to the operation team of the TRIGA IPR-R1 Reactor for their help during the experiments.

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